April 24, 2002

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: INCREASE IN LICENSED POWER LEVEL (TAC NO. MB0789)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 244 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Operating License and Technical Specifications (TSs) in response to your application dated December 19, 2000, as supplemented by letters dated May 30, June 20, 26 (two letters), 27, and 28, July 3 and 24 (two letters), August 7, 13, 21, 23, and 30, September 14, October 1, 12 (two letters), 17, 30 (two letters), and 31, November 9, 16 (three letters), and 17, and December 5, 6 (two letters), 10, and 20, 2001, and January 14, 15, and 31, February 7 (two letters), and March 1, 2002.

The amendment allows an increase in the maximum authorized reactor core power level from 2815 megawatts thermal (MWt) to 3026 MWt, which represents a power increase of about 7.5 percent and is considered to be an extended power uprate. Also, operation at the uprated power requested by the proposed amendment results in increases in dose consequences for certain postulated accidents considered in the accident analyses in the Safety Analysis Report; however, the doses remain within the regulatory limits. In addition, although unrelated to the proposed power uprate, the proposed amendment clarifies portions of the control element assembly TSs.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 244 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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Distribution - See next page

TS Pages: ML021150408 **NRR-100** **PKG.: ML021140674**
Accession No.: ML021130826 **NRR-058** ***see previous concurrence**

OFFICE	PDIV-1/PM	PDIV-1/LA	TECH ED	OGC	PDIV-1/SC
NAME	TAlexion	DJohnson	PGarrity*	JHeck*	RGramm
DATE	04/19/02	4/19/02	04/08/02	04/12/02	4/19/02

OFFICE	PDIV/D	DLPM/D	NRR/D
NAME	SRichards*	TMarsh for JZwolinski	JJohnson for SCollins BWS
DATE	04/09/02	4/19/02	3/23/02

The following technical branches provided safety evaluation input by memos dated*

OFFICE	SRXB/SC	SPLB/(A)SC	EMEB/SC	EMCB/SC	SPSB/BC
NAME	FAkstulewicz[1]	HWalker	KManoly	KWichman[2]	RBarrett
DATE*	8/23&12/4&26/01	11/27/01	12/04/01	10/15/01	01/08/02

OFFICE	IOLB/SC	IQPB/BC	EEIB/SC	EEIB/SC	SPSB/SC
NAME	DTrimble	TQuay[3]	CHolden	EMarinos	MReinhart
DATE*	07/19/01	10/17/01	05/10&11/29/01	10/03/01	02/05/02

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[1] Also concurred on final SE per e-mail dated 03/25/02.

[2] M. Mitchell (EMEB/(A)SC) provided additional SE input per e-mail dated 02/25/02.

[3] SE input revised by D. Thatcher (IEHB/SC) per e-mail dated 04/18/02.

Distribution for Letter dated: April 24, 2002

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: INCREASE IN LICENSED POWER LEVEL (TAC NO. MB0789)

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 244
License No. NPF-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated December 19, 2000, as supplemented by letters dated May 30, June 20, 26 (two letters), 27, and 28, July 3 and 24 (two letters), August 7, 13, 21, 23, and 30, September 14, October 1, 12 (two letters), 17, 30 (two letters), and 31, November 9, 16 (three letters), and 17, and December 5, 6 (two letters), 10, and 20, 2001, and January 14, 15, and 31, February 7 (two letters), and March 1, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 244, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   In addition, the license is amended to revise paragraph 2.C.(1) of Facility Operating License No. NPF-6 to reflect the new authorized reactor core power level. The license is also amended to authorize revision of the Safety Analysis Report regarding the dose consequences for certain postulated accidents as set forth in the application for amendment by Entergy Operations, Inc., dated December 19, 2000, as supplemented by letters dated May 30, June 20, 26 (two letters), 27, and 28, July 3 and 24 (two letters), August 7, 13, 21, 23, and 30, September 14, October 1, 12 (two letters), 17, 30 (two letters), and 31, November 9, 16 (three letters), and 17, and December 5, 6 (two letters), 10, and 20, 2001, and January 14, 15, and 31, February 7 (two letters), and March 1, 2002, and evaluated in the staff's safety evaluation enclosed with this amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to the commencement of heatup from refueling outage 2R15.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jon R. Johnson, Acting Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License and Technical Specifications

Date of Issuance: April 24, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 244

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the Operating License and Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

License Pages

Remove	Insert
3	3

Technical Specification Pages

Remove	Insert
1-1	1-1
2-5	2-5
3/4 1-17	3/4 1-17
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-25	3/4 1-25
3/4 1-26	3/4 1-26
3/4 3-5b	3/4 3-5b
3/4 3-5c	3/4 3-5c
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 5-7	3/4 5-7
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
6-21a	6-21a
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 3/4 2-1
B 3/4 5-3	B 3/4 5-3
B 3/4 7-1	B 3/4 7-1

ARKANSAS NUCLEAR ONE, UNIT 2
SAFETY EVALUATION FOR EXTENDED POWER UPRATE

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 244 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated December 19, 2000, as supplemented by letters dated May 30, June 20, 26 (two letters), 27, and 28, July 3 and 24 (two letters), August 7, 13, 21, 23, and 30, September 14, October 1, 12 (two letters), 17, 30 (two letters), and 31, November 9, 16 (three letters), and 17, and December 5, 6 (two letters), 10, and 20, 2001, and January 14, 15, and 31, February 7 (two letters), and March 1, 2002, Entergy Operations, Inc. (Entergy or the licensee), submitted a request for changes to the Arkansas Nuclear One (ANO), Unit No. 2 (ANO-2), Operating License (OL) and Technical Specifications (TSs). The proposed amendment would allow an increase in the maximum authorized reactor core power level from 2815 megawatts thermal (MWt) to 3026 MWt, which represents a power increase of about 7.5 percent (%) and is considered to be an extended power uprate. The proposed amendment would change the OL and certain TSs to provide for implementing uprated power operation. Also, operation at the uprated power requested by the proposed amendment would result in increases in dose consequences for certain postulated accidents considered in the accident analyses in the Safety Analysis Report (SAR); however, the doses would remain within the regulatory limits. In addition, although unrelated to the proposed power uprate, the proposed amendment would clarify portions of the control element assembly (CEA) TSs.

2.0 BACKGROUND

Approval of this application would authorize operation of ANO-2 at a Nuclear Steam Supply System (NSSS) power level up to 3026 MWt, which represents a 7.5% increase above the currently licensed core power rating of 2815 MWt. The proposed change would increase the unit's best estimate gross electrical output from about 958 megawatts electric (MWe) to about 1065 MWe.

The proposed power uprate represents the first power uprate for ANO-2 since original issuance of the OL. The 7.5% uprate was selected based on several factors. Since the ANO-2 steam generators (SGs) required replacement in the fall of 2000, due to various corrosion-related phenomena that occurred over the years of operation, the licensee made an economic decision to design the replacement SGs (RSGs) to accommodate an increase in rated thermal power (RTP). It was determined that a power uprate of at least 6.5% would be required in order to recover the capital investment of larger RSGs and other modifications that would be necessitated by the installation of larger RSGs. Scoping efforts were then initiated by the licensee to explore whether an uprate greater than 6.5% was possible based on five criteria:

- satisfactory safety analysis results,
- satisfactory margins on all safety-related systems, structures and components,
- satisfactory margins for reactor vessel (RV) head Alloy 600 nozzles,
- acceptable additional cost above the cost to achieve a 6.5% uprate, and
- the ability of the RSGs to support a higher uprate.

Based on the above criteria and the physical limitations of the RSGs, a 7.5% power uprate was determined by the licensee to be the optimum level.

The staff has reviewed the licensee's application to ensure that it meets the applicable regulations. The staff has also reviewed the licensee's analyses to ensure that the licensee performed the analyses using codes and methodologies that the staff finds acceptable. In addition, the staff applied the experience it has gained from performing previous power uprate reviews on other plants, and performed independent calculations and a site audit in selected areas. These areas include: 1) limited analyses related to boron precipitation and long-term cooling to assure that the time contained in the licensee's procedures for operator actions to switchover from cold-leg injection to hot-leg injection was acceptable, 2) independent calculations (by a Nuclear Regulatory Commission's (NRC or Commission) contractor) of the peak containment pressures and temperatures following postulated accidents, 3) independent calculations on the susceptibility to vessel head penetration cracking, 4) independent calculations of the atmospheric dispersion estimates (which input into the radiological dose assessments) and independent dose assessments following postulated accidents, and 5) a site audit of the licensee's risk evaluation, which included reviewing the human reliability analysis methodology and selected human error probability calculations. The staff's evaluation of the licensee's request for ANO-2 follows.

3.0 NUCLEAR STEAM SUPPLY SYSTEM, ACCIDENT ANALYSIS, AND OTHER DESIGN-BASIS EVALUATIONS

3.1 Nuclear Steam Supply System Parameters

In its Power Uprate Licensing Report (PULR), the licensee provided a summary of key operating conditions and corresponding analytical assumptions. The key parameters include reactor power, axial shape index, RV inlet temperature, pressurizer pressure, reactor coolant flow rate, feedwater temperature, steam/feedwater flow rate, and steam pressure. These key plant parameters are established for the evaluations of power uprate with the RSGs. Cycle 16 will be the first cycle at the uprated power conditions. The values of these parameters are demonstrated acceptable by various safety analyses using these parameters. The staff finds that these power uprate parameters are acceptable based on the acceptable results of the accident analysis evaluation addressed in Section 3.6 below.

3.2 Reactor Coolant System (RCS) Operating Conditions

The RCS operating conditions are changed slightly at uprated power. The RCS flow rate is about the same and the RCS temperatures are increased. The higher temperatures associated with the power uprate are still within the bounds of the original design temperature of 650 ^{o}F for the RCS and 700 ^{o}F for the pressurizer, and an original design hot-leg temperature (T-hot) of 612 ^{o}F. Sufficient core cooling under power uprate conditions is verified by various plant transient and safety analyses. The results of the licensee's calculations have concluded that a natural circulation cooldown without drawing a steam bubble in the RV can be achieved with power uprate operating conditions. The staff finds that the changes of RCS operating parameters associated with power uprate are acceptable based on the results of the accident analysis evaluation addressed in Section 3.6 below.

3.3 Safety Injection System

The adequacy of the safety injection system during the injection and sump recirculating phases is verified by various safety analyses performed in support of the power uprate. There are no system modifications required to support power uprate. The staff agrees with the licensee's assessment based on the acceptable results of the accident analysis evaluation addressed in Section 3.6 below.

3.4 Shutdown Cooling System

The licensee has verified the ability of the shutdown cooling system to achieve cold shutdown within 36 hours, and to maintain refueling temperatures and a uniform boron concentration in the RCS, under the power uprate conditions. Based on its evaluation, the licensee has concluded that system modifications are not required to accommodate the power uprate. The staff has reviewed the licensee's submittal and finds the licensee's assessment acceptable.

3.5 Nuclear Steam Supply System Design Transients

In its PULR, the licensee has redefined and evaluated the NSSS design transients to account for any impacts of the RSGs and power uprate. The methodology for defining the NSSS design transients ensures that they bound both normal and abnormal plant operations. The number of occurrences of any given transient selected for design purposes exceeds the expected number over the life of the plant. The intent is to ensure that no RCS component is stressed above the allowable limits as described in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Actual plant data was considered in estimating the design frequency of occurrences for some events.

The licensee provided a tabulation which shows comparison of design transients that are different between the RSGs and original SGs (OSGs). The number of design transients for the RSGs was determined to account for the design life and thermal hydraulic performance of the RSGs under power uprate conditions. The number of design transients for the loss-of-feedwater event is changed from 8 to 20 for the RSGs. This change in the conservative direction is based upon historical plant performance at ANO-2. The number of hydrostatic tests was reduced from 10 to 1 due to the change of ASME Code requirements. Also, the SG leak testing mode is changed for the RSGs to reflect the current methodology used to test for leakage.

As discussed in its original request and response to staff questions, the licensee evaluated the effect of the power uprate on the RSG leak testing at ANO-2. The licensee indicated that following installation of the RSGs, leak testing will be performed in accordance with ASME Section XI Code Case N-416-1, "Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Item by Welding, Class 1, 2 and 3 Section XI, Division 1." With the exception of the leak tests listed in Table 6-6 of the December 19, 2000, application, leak tests are performed during plant operation at the normal operating pressure. The fatigue considerations of these leakage tests are accounted for in the plant heatup and cooldown transients, and accordingly, are not required in conjunction with the leak tests. However, two RSG shop hydrostatic tests (one each for the primary and secondary sides) and the anticipated tube leak pressure tests as listed in Table 6-6 of the application have been considered in the existing fatigue analysis. The primary and secondary shop hydrostatic tests were performed using the system design pressures, which will not change as a result of power uprate. The tube leak tests are performed during shutdown low pressure conditions and therefore, are not affected by the power uprate. Therefore, the licensee concluded that the ANO-2 leak testing transients will not be affected by the proposed power uprate and that the existing fatigue analysis associated with the leak testing remains unchanged for the uprate operation. The staff has reviewed the licensee’s submittal and finds its conclusion acceptable.

3.6 Accident Analysis Evaluation

In support of this power uprate, the licensee re-evaluated the safety analyses for the operation of ANO-2 at a RTP of 3026 MWt with the power measurement uncertainty of 2%. The changes of NSSS parameters discussed previously in this safety evaluation are used in the analyses to support power uprate. The licensee performed the majority of the uprate analyses and evaluations in accordance with the current ANO-2 licensing basis methodologies.

Table 7.3.0.1-1 of the application presents the key parameters assumed in the transient analyses. Specific initial conditions for each event are listed in that events’ section. Events were evaluated to determine the effects of power uprate and bounding parameters. For those events for which a detailed analysis was performed (see Table 7.3.0-1), the initial core power was assumed to be 3026 MWt.

Many of the analyses performed by ANO-2 used physics data that is expected to bound future core designs. Physics data such as moderator temperature coefficient (MTC), fuel temperature coefficient (Doppler coefficients or curves), and delayed neutron fractions, are typical of core physics parameters that are considered on a cycle-by-cycle basis. A set of these physics parameters were presented in the application. The staff will discuss the specific analyses in the appropriate safety analysis sections of this safety evaluation. The radiological analysis evaluation of selected postulated accidents is discussed separately in Section 7.0 of this safety evaluation.

The ANO-2 power uprate analyses and evaluations were performed using the following guidelines: 1) Westinghouse Electric Company (Westinghouse) Topical Report WCAP-10263, “A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Plant,” dated January 1983; 2) General Electric Company (General Electric or GE) Topical Report NEDC-31897P-A, “Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,” May 1992; and 3) SECY-97-042, Section 3, “Power Uprate Review Process.”

The licensee has provided tabulations to list: 1) all computer codes and methodologies applied to each of the transient and accident analyses, 2) the restrictions and limitations identified in the NRC safety evaluation for each of the topical reports relative to computer codes and methodologies, and 3) discussion of how these applicable restrictions and limitations of the computer codes and methodologies specified in the staff safety evaluations are complied with in the ANO-2 power uprate analyses. The staff finds that the licensee has performed a detailed study to confirm that the computer codes and methodologies applied in its power uprate analyses are adequate.

3.6.1 Loss-of-Coolant Accident (LOCA) Analysis

3.6.1.1 Large-Break Loss-of-Coolant Accident (LBLOCA) and Small-Break Loss-of-Coolant Accident (SBLOCA)

The licensee performed ANO-2 LBLOCA and SBLOCA analyses assuming 102% (3087 MWt) of the proposed 7.5% uprated power (3026 MWt). The licensee performed the LBLOCA and SBLOCA analyses using methodologies presently approved for ANO-2 licensing applications. The licensee reported the results of its LBLOCA and SBLOCA analyses in its application dated December 19, 2000, proposing the ANO-2 power uprate.

The licensee showed that the LBLOCA and SBLOCA analysis methodologies presently approved for ANO-2 continue to apply specifically to the ANO-2 plant, in a letter dated October 31, 2001, by providing a statement that the licensee and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature (PCT)-sensitive parameters bound the as-operated plant values for those parameters. The staff also considered the continued applicability of the methodologies' constituent models at the uprated power. This was of particular concern for the SBLOCA methodology. The licensee's provision of the ongoing processes stated above resolved this concern by assuring continued conservatism of the ANO-2 SBLOCA methodology at the 7.5% uprated power (3026 MWt).

The licensee's December 19, 2000, application also provided the LBLOCA and SBLOCA analysis results for operation of ANO-2 at 3026 MWt. The licensee's results of the analyses are PCTs of 2154 ^{o}F for LBLOCA and 2090 ^{o}F for SBLOCA, maximum local cladding oxidation values of 7.8% for LBLOCA and 12.5% for SBLOCA, and maximum core-wide oxidation values of less than 0.99% for LBLOCA and 0.73% for SBLOCA. These are within the criteria given in 10 CFR 50.46(b)(1-3). From these results the licensee has concluded that the core geometry will remain coolable as required by 10 CFR 50.46(b)(4).

Based on the licensee's demonstration that the LBLOCA and SBLOCA analysis methodologies presently approved for ANO-2 continue to apply specifically to the ANO-2 plant and that the analyses assumed an initial power level of 1.02 times the proposed licensed power level, the staff finds that the LOCA analysis methodologies presently approved for ANO-2 continue to apply to the ANO-2 plant and are suitable for inclusion in plant licensing documentation, including the TSs and the core operating limits report (COLR).

3.6.1.2 Emergency Core Cooling System (ECCS) Switchover and Post-Loss-of-Coolant Accident Long-Term Cooling (LTC)

Pursuant to 10 CFR 50.46(b)(5), "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Sections 3.6.1.2.1 and 3.6.1.2.2 of this safety evaluation discuss how this requirement is met.

3.6.1.2.1 Emergency Core Cooling System Switchover from Cold Leg Injection from the Refueling Water Tank (RWT) to Cold Leg Injection from the Emergency Core Cooling System Sump

When a sufficient amount of water has been injected by the ECCS to satisfy various design requirements, including providing enough water in the ECCS sump to meet ECCS pump net positive suction head requirements, but prior to depletion of the RWT inventory, the ECCS suction is switched from the RWT to the ECCS containment sump. The increased residual heat due to the higher (uprated) power could impact the timeliness of actions necessary to effect the switchover of the injection source. In a supplemental letter dated October 17, 2001, the licensee stated that the ANO-2 switchover is automatic and that ECCS flow to the reactor core is not interrupted during this switchover. Therefore, the core cooling demonstrated by the LBLOCA and SBLOCA analyses is maintained through the switchover, and the timeliness of actions to make the switchover occur is not an issue. The staff, therefore, finds this acceptable.

3.6.1.2.2 Post-Loss-of-Coolant Accident Long-Term Cooling Including Hot Leg Injection

LTC requirements following a LOCA are established by 10 CFR 50.46(b)(5). One aspect of LTC is to ensure that boric acid (H_3BO_3) accumulation will not prevent core cooling, an aspect potentially affected by the proposed power uprate. Consequently, the staff audited the licensee's H_3BO_3 evaluation model (EM) for analysis of H_3BO_3 accumulation characteristics following a LOCA.

The licensee's response to Question 18 in its October 17, 2001, supplemental letter, states that the analysis used for the power uprate application "...uses the Westinghouse boric acid precipitation evaluation model...from CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model, June 1980,..." and that this analysis "...replaces the LTC analysis documented in...the...SAR." The October 17, 2001, supplemental letter states that the SAR analysis and the power uprate analysis both contain the following assumptions:

1. No credit is taken for subcooling of the safety injection flow.

2. No credit is taken for entrained liquid leaving the core (only steam leaves the core).

3. No credit is taken for boric acid volatility.

4. Prior to the initiation of simultaneous hot and cold side injection, the only injection into the reactor vessel credited is that which is required to replace boil-off.

5. Maximum boric acid concentrations are represented for all sources of boric acid.

6. Maximum initial liquid volumes are represented for all sources of boric acid.

7. No credit is taken for increased boric acid solubility due to boiling point elevation.

8. Credit is taken for mixing of liquid in the core and lower plenum. This credit is based on the results of a post-LOCA boric acid concentration test.

Items 1 through 7 are consistent with assumptions traditionally accepted by the staff for licensee H_3BO_3 accumulation analyses during LTC. The staff finds them acceptable for the proposed power uprate analysis. Item 8 has not been satisfactorily substantiated, as discussed below.

CENPD-254-P-A does not specify the well-mixed volume where H_3BO_3 accumulates; however, a footnote on page 20 in CENPD-254, Amendment 1, states, "The BORON code maintains a constant input specified vessel mixing volume that is conservatively determined." The licensee's October 17, 2001, supplement describes the lower plenum as part of the well-mixed volume and states that the SAR and power uprate volumes are different. The staff's examination of the two volumes indicated that the power uprate volume is more conservative than the SAR volume. The licensee justifies the well-mixed lower plenum assumption on the basis of a test; however, no test information, references, or supporting analyses are provided to justify crediting the lower plenum. The staff briefly audited the vendor's proprietary test reports in Westinghouse's Rockville, Maryland office on December 6, 2001, and determined there were no scaling or other analyses to support quantitative lower plenum mixing conclusions. The staff found that the licensee has not justified existence of a well-mixed volume below the elevation of the top of the core support plate.

The October 17, 2001, supplemental letter states that the SAR analysis assumes a H_3BO_3 solubility limit of 32 weight percent (wt%), corresponding to a 20 pounds per square inch, absolute (psia) pressure, and the power uprate analysis uses 27.6 wt%, corresponding to a 14.7 psia pressure. In this respect, the power uprate analysis is clearly more conservative than the SAR analysis. The licensee's analysis has no allowance for uncertainty. The staff notes, however, that it is consistent with CENPD-254-P-A.

The October 17, 2001, supplemental letter also states that the decay heat generation rate assumed in the power uprate analysis uses a multiplier of 1.2 for the first 1000 seconds following initiation of the LOCA, and uses a factor of 1.1 thereafter. This is consistent with the approved CENPD-254-P-A assumption, as stated on page C-5. Although neither reference provides a direct statement regarding the irradiation time, page C-6 of CENPD-254-P-A provides the decay heat energy equations. An audit of the equation for the first 10^5 seconds showed it to be reasonably consistent with the stated power uprate assumption for an extended irradiation time.

In summary, assumptions 1 through 7, above, are acceptable for the power uprate analysis. Inclusion of the lower plenum in the assumed well-mixed volume and use of the assumed H_3BO_3 concentration limit without a 4 wt% margin have not been acceptably justified in the licensee’s recent supplemental letters. The staff notes, however, that the NRC did not object to the use of the lower plenum mixing assumption in the SAR analysis at the time of licensing, nor did it object to omission of the 4 wt% margin in CENPD-254-P-A.

Additional insight can be obtained by comparing the licensee’s SAR and power uprate analysis predictions. Some of the licensee’s H_3BO_3 accumulation analysis predictions are provided in Table 3.1, below. Items 1 and 2 of Table 3.1 provide a direct comparison of the SAR and power uprate analysis predictions. Figure 1 of the licensee's December 20, 2001, supplemental letter illustrates a steep initial increase rate in H_3BO_3 concentration for Item 2 that does not appear in Item 1. This dominates the change from 9.5 hours to 2.4 hours. It is due to the power uprate analysis assumption that the makeup tank is injected into the core until it is empty. The behavior apparently is not predicted by the SAR analysis because the makeup tanks are not treated this way. (The licensee’s records do not appear sufficient to establish this SAR analysis assumption.) The claimed reduction from 9.5 hours to 2.4 hours is consistent with a conclusion that the power uprate analysis is conservative when compared to the existing licensing-basis SAR analysis. In Item 5, the 7.3-hour prediction is consistent with the 2.4-hour prediction in Item 2 because the makeup tank concentration is reduced from 12 wt% to 3.5 wt% H_3BO_3, which reduces the initial slope. In addition, the 7.3-hour prediction in Item 5, which is for the proposed power uprate conditions, supports a conclusion that the power uprate analysis appears to be conservative if compared to a SAR analysis prediction.

Table 3.1 - Predicted Limiting H_3BO_3 Accumulation Times Without Hot-Leg Injection			
Item	Description	Time (hours)	Reference
1	SAR method for Cycle 1 with 32 wt% H_3BO_3 for present licensed power.	9.5*	1
2	Power uprate method (CENPD-254) substituted for SAR method in Item 1.	2.4*	1
3	Item 2 with proposed power uprate but no other changes.	< 2.4*	1
4	Operator action time stated in SAR that occurs with < 20 wt% H_3BO_3.	4**	2
5	Power uprate method for proposed power uprate operation. Includes reduction of H_3BO_3 makeup tank H_3BO_3 concentration from 12 wt% to 3.5 wt% (Amendment 82), reduction of allowable H_3BO_3 from 32 wt% to 27.6 wt%, and changes in the assumed well-mixed volume.	7.3*	1, 3
6	Assumed initiation of hot-leg injection for further analyses for Item 5 conditions. Maximum H_3BO_3 concentration is 23.3 wt% at 5.9 hours.	5**	1, 3
7	Current emergency operating procedure (EOP) initiation.	2 to 4	1, 2
*Time to reach H_3BO_3 saturation condition. **Time at which operator is to take action. Reference 1. December 20, 2001, supplemental letter. Reference 2. SAR Amendment 16. Reference 3. December 19, 2000, application.			

Further insight can be obtained by comparing the licensee's predictions with those used by another licensee using a different model in a power uprate request. Table 3.2, below, provides a comparison of the Byron Station and Braidwood Station (Byron/Braidwood) characteristics with the licensee's. (The H_3BO_3 accumulation characteristics associated with the Byron/Braidwood power uprate were conditionally approved by the staff on May 4, 2001). An immediate conclusion is that the ANO-2 methodology predicts a shorter available time than did the Byron/Braidwood methodology for the respective plants. One may approximately "correct" for the differences listed in Table 3.2 by ratioing values to obtain an ANO-2 prediction from the Byron/Braidwood value of 8.53 hours. The staff-generated value via ratioing is 9.1 hours, which is greater than the 7.3 hours predicted by the power uprate analysis. The staff notes that this comparison is a gross evaluation. It is not consistent with the respective plant analyses. For example, the core region volume and the upper plenum volume below the elevation of the hot-leg nozzles is used in the comparison. These volumes are not consistent with the licensee's assumed volumes. In addition, the staff did not find the Byron/Braidwood analysis to be acceptable, although it did state it was confident that the licensee could establish acceptable results when the analysis was suitably modified. Nevertheless, even though this comparison is not rigorous, it provides a sufficient basis to find that the licensee's power uprate analysis prediction is a reasonable estimate.

Table 3.2 - Comparison of Byron/Braidwood and ANO-2 Characteristics			
Item	Characteristic	Byron/Braidwood 5% power uprate (May 4, 2001)	ANO-2
1	Time to reach H_3BO_3 saturation (hours).	8.53	7.3
2	Power (MWt).	3587	3026
3	Decay heat generation rate multiplier (dimensionless).	1	1.1
4	Assumed H_3BO_3 saturation limit (wt%).	23.53	27.6
5	Assumed ECCS enthalpy change (BTU/lb boiled).	1013.05	970.86
6	Core plus upper plenum volume below hot leg (ft^3).	1072*	940
*Value is from NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon Nuclear Power Plant, Unit 2, April 10, 1987," June 1987.			

In summary, while the staff cannot endorse the licensee's evaluation, the staff believes there is sufficient basis to approve the license amendment. This basis includes:

1) There is a low probability that conditions leading to significant H_3BO_3 accumulation will be encountered.

2) A partial quantification of approved EM conservatisms shows that there are several conservatisms that tend to compensate for the effect of issues identified in this safety evaluation.

3) As identified above, the proposed power uprate analysis is more conservative than the SAR analysis-of-record (AOR).

4) The licensee's predictions are reasonable when approximately compared to those for a different plant using a different analysis.

5) The staff is confident that the licensee, using a fully substantiated model, could support the 2 hours to 4 hours it states is used in its EOPs.

Consequently, the staff does not consider the outstanding issues to be a significant safety concern, and the staff will accept the licensee's EOPs' 2 hour to 4 hour instruction for initiation of hot-leg injection.

The staff, in its evaluation of CENPD-254-P, stated that "Should NRC criteria or regulations change such that our conclusion concerning CENPD-254-P are invalidated, you will be notified and given an opportunity to revise and resubmit your topical report should you so desire." The staff intends to resolve the meaning of conservatism in determination of vessel mixing volume and the 4 wt% H_3BO_3 margin question on a generic basis consistent with its CENPD-254-P safety evaluation statement.

3.6.2 Non-Loss-of-Coolant Accident Events Analysis

The licensee has analyzed the design bases non-LOCA events to support the power uprate. The analyses generally use the NRC-approved methods. In conjunction with the analyses, the core operating limits supervisory system (COLSS) is used to monitor the departure from nucleate boiling (DNB) ratio (DNBR) and the linear heat rate (LHR) margins, and the core protection calculator system (CPCS) is used to protect against violation of the DNB and LHR specified acceptable fuel design limits (SAFDLs) during the anticipated operational occurrences. The fuel peaking factors are determined in the fuel reload core design to ensure that adequate operating margin is maintained. However, a change of analysis methodology was made in the analysis for a feedwater line break (FWLB). The methodology was slightly modified to credit the reactor trip on low water level signal from the ruptured SG in the FWLB accident analysis. The staff evaluation of this modified methodology is addressed in Section 3.6.2.14 below.

The acceptance criteria for the anticipatory operational occurrences analyzed are that the calculated minimum DNBR remains greater than the safety limit and the peak RCS pressure remains less than the safety limit of 110% of design pressure (i.e., 2750 psia). The licensee analyzed the design transients with the NRC-approved methods: the COLSS and the CPCS to monitor the DNBR and the LHR margins.

3.6.2.1 Uncontrolled Control Element Assembly Withdrawal from Subcritical Conditions

The uncontrolled CEA withdrawal from a subcritical condition was analyzed using the approved computer codes addressed in Section 3.6 of this safety evaluation. The input parameters and the associated uncertainties assumed in the analysis were consistent with the current analysis in the SAR. The analysis was performed with acceptable results at the uprated conditions and increased RCS flow. This was demonstrated for the limiting subcritical CEA bank withdrawal event, where the peak fuel temperature was found to be well below the centerline melt temperature. The peak heat flux resulted in a minimum DNBR greater than 1.25. Since this is not a limiting peak RCS pressure event, the effect of the power uprate on this event is acceptable.

3.6.2.2 Uncontrolled Control Element Assembly Withdrawal from Critical Conditions

To analyze this event, the licensee used the methodology documented in the AOR. The methodology utilizes the CENTS computer code for the transient analysis simulation, and uses the CETOP code to determine the minimum DNBR. Two sets of cases were analyzed by the licensee for this event: one initiated at hot full power (HFP) and one initiated at hot zero power (HZP). Both of these cases were analyzed using end-of-cycle (EOC) kinetics. The simulated event assumed the maximum reactivity addition rate based on a CEA withdrawal speed of 30 inches per minute.

At ANO-2, a CPCS low DNBR trip, a CPCS high local power density trip, or a CPCS variable overpower trip will terminate the sequential CEA withdrawal events. The CPCS has dynamic compensation lead-lag filters that project increases in core heat flux and core power. These dynamic compensation filters in conjunction with static power correction factors ensure that the CEA withdrawal transients are terminated before the SAFDLs are violated.

For the limiting HFP CEA bank withdrawal event, the peak core power resulted in a peak LHR of less than 21 kilowatts per foot. For the limiting HZP CEA bank withdrawal event, the peak core power resulted in fuel temperatures well below the centerline melt. The licensee's analysis also identified that the limiting fuel centerline temperature occurred at the HZP with the EOC kinetics. For both limiting cases, the calculated DNBRs and fuel centerline temperatures are within the safety limits, thus, ensuring no fuel failures during this event. The peak heat flux results in a minimum DNBR greater than 1.25. Since these are not limiting peak RCS pressure events, there is no violation of the SAFDLs for both the HFP and the HZP events. In addition, the results meet the acceptance criteria of Standard Review Plan (SRP) 15.4.8 and, therefore, are acceptable.

3.6.2.3 Control Element Assembly Misoperation

Control rod misoperation events were assumed to initiate from a dropped full-length rod cluster control assembly (RCCA), and a statistically misaligned RCCA. The core protection calculator (CPC) and the core element assembly calculator algorithms detect and compensate for the effect of CEA misoperation on the core power distribution by providing heat flux and radial peaking factor penalties to the online DNBR and LHR calculations. The licensee analyzed the control rod misoperation events using the NRC-approved computer codes, ROCS and CETOP, for the transient and DNBR calculations. The input parameters and the associated uncertainties assumed in the analysis were consistent with the increase in power and the current analysis in the SAR.

The analysis shows that the single CEA drop event radial peaking factor limits were determined to ensure that the DNBR and LHR SAFDLs are not exceeded. The staff finds that the results have met the SRP 15.4.3 acceptance criteria and concludes the results are acceptable.

3.6.2.4 Idle Loop Startup

The current TSs at ANO-2 preclude power operation with an inactive loop. Therefore, this event is not analyzed for power uprate.

3.6.2.5 Uncontrolled Boron Dilution Incident

In SRP 15.4.6, the staff requires that at least 15 minutes are available from the time the operator becomes aware of the unplanned boron dilution event to the time a total loss of shutdown margin occurs during Modes 1 through 5. A warning time of 30 minutes is required during Mode 6. The licensee analyzed the uncontrolled boron dilution events during Modes 1 through 6 using methods consistent with the current analysis in the SAR. The input parameters and the associated uncertainties used in the analysis were consistent with the AOR and power uprate.

The dilution during refueling analysis assumed a boron concentration consistent with the TS limit (K-effective = 0.95), and the critical boron concentration/inverse boron worth (CBC/IBW) limit line as presented in Figure 7.3.4-1 of the application was based on a conservative value of 31 minutes from alarms to loss of shutdown margin.

Dilution during cold shutdown with the RCS filled was also analyzed with the results indicating that if the count rate monitors are operable and no CEAs are withdrawn, the count rate monitors will alarm, or if the count rate monitors are not operable and the CEAs are withdrawn, the high logarithmic power trip will alert the operator more than 15 minutes before the loss of all shutdown margin. The licensee stated in its application that the operators will be alerted to the event with more than the minimum 15 minutes of response time available. In addition, the CBC/IBW limit lines present in Figures 7.3.4-3 and 7.3.4-4 of the application, for alarms inoperable and operable, respectively, were based on 16 minutes from alarms to loss of shutdown margin.

3.6.2.6 Total and Partial Loss of Forced Reactor Coolant Flow

A total loss of forced reactor coolant flow can result from an occurrence of a mechanical or electrical failure. Failure of an electrical system to all four reactor coolant pumps (RCPs) would cause a total loss of forced reactor coolant flow. The licensee analyzed two complete loss of forced reactor coolant flow cases at power uprate conditions using methods that the staff has previously approved. They are: 1) minimum subcooling case, and 2) maximum subcooling case. The licensee's analysis of case 2 provides more limiting results because maximum subcooling conditions would result in the least amount of negative reactivity inserted due to generation of voids in the core. In this manner, the system undergoes the greatest amount of thermal margin degradation following the RCP coastdown. The results of this bounding analysis show that the minimum DNBR is greater than the safety analysis limit of 1.25.

A partial loss of forced reactor coolant flow was not analyzed for power uprate since the reactor coolant flow coastdown under these conditions is less severe than that for a total loss of reactor coolant forced flow. The staff has reviewed the assumptions and the results of the licensee's analysis, and finds that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.6.2.7 Loss of Reactor Coolant Flow Resulting from a Pump Shaft Seizure

The licensee analyzed the single RCP shaft seizure accident at power uprated conditions using methods that the staff has previously approved. The results of the licensee's analysis show that the peak RCS pressure remains within 110% of its design pressure. The present licensing basis for the pump shaft seizure uses an estimate of 14% failed fuel rods to evaluate the radiological consequences. If the number of failed fuel rods in this event remains less than 14%, then the radiological consequences will be acceptable since they would be bounded by the current radiological assessment for this event. The licensee stated that the amount of fuel failure will be verified in the Cycle 16 thermal-hydraulic analysis. During the current cycle, the predicted fuel damage was 2.9%. The staff believes that the final calculated amount of fuel failure for this event in Cycle 16 will be within the current assumed number of 14%. This is because the licensee's analysis of this event did not assume a concurrent loss-of-offsite power (LOOP) which will result in less degradation of thermal margin during this event. Upon completion of the final Cycle 16 reload analyses, the licensee agreed to provide to the NRC, in writing, the actual calculated amount of fuel failure to confirm that the predicted fuel failure for this event is less than 14%. By supplemental letter dated January 15, 2002, the licensee stated that the Cycle 16 reload efforts on the RCP shaft seizure accident were completed and that the maximum failed fuel percentage was calculated to be 4.95%.

The staff noticed that the licensee did not assume a concurrent LOOP in the analysis of this accident. However, the licensee's approach is consistent with its current licensing basis and, therefore, acceptable. The staff has reviewed the other assumptions and the results of the licensee's analysis, and concluded that the other assumptions used in this analysis are conservative. Therefore, the staff finds the licensee's assessment acceptable.

3.6.2.8 Loss of External Load and/or Turbine Trip

The loss of external load and/or turbine trip event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222 dated September 29, 2000. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.9 Loss of Normal Feedwater

The loss of normal feedwater event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.10 Loss of all Normal and Preferred Alternating Current (AC) Power to the Station Auxiliaries

The loss of all normal and preferred AC power to the station auxiliaries event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.11 Excess Heat Removal Due to Feedwater System Malfunction

The excess heat removal due to feedwater system malfunction event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.12 Excess Heat Removal Due to Main Steam System Valve Malfunction

The excess heat removal due to main steam system valve malfunction event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.13 Steam Line Break Accident With or Without Concurrent Loss of Alternating Current (AC) Power

The steam line break accident with or without concurrent loss of AC power was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.14 Feedwater Line Break Accident

The licensee has analyzed the feedwater system pipe break accident at the power uprate conditions. The methodology used for the licensee's analysis is slightly changed from that used in the previous analysis. The previous analysis determined the limiting FWLB size by combining a simultaneous high pressurizer pressure trip and low SG level trip on the intact SG. The new analysis credits a low SG level trip on the affected SG that will occur with at least 40,000 pounds-mass (lbm) liquid inventory remaining in the SG. The limiting FWLB size is calculated based on a simultaneous high pressurizer pressure trip and low SG level trip on the affected SG at 40,000 lbm. In response to the staff's question regarding the reliability of the reactor trip on low water level from the affected SG, the licensee stated that the instrument uncertainty calculations have taken into consideration the SG conditions when determining the mass of inventory in the SG at the time of trip. The blowdown effects of density changes and velocities following a FWLB have been accounted for. An inventory of 40,000 lbm credited in the analysis is conservative with respect to the approximate 78,000 lbm at the low level trip set point credited in the loss of feedwater analysis. Credit for low level indication during a FWLB on the affected SG is similar to the credit taken by Westinghouse plants as presented in WCAP 9230, "Report on the Consequences of a Postulated Main Feedline Rupture," and WCAP 9236, "NOTRUMP: A Nodal Transient Steam Generator and General Network Code." The RSGs for ANO-2 are Westinghouse-designed SGs. The 40,000 lbm was determined using the NOTRUMP code. This 40,000 lbm assumption was then used in the CENP CENT code for determination of the effects on the RCS. The staff has reviewed the licensee's submittal and finds the revised methodology acceptable.

The results of the analysis demonstrate that the peak primary and secondary pressures are maintained below 110% of respective design pressures. Also, it is demonstrated that the assumed emergency feedwater system (EFWS) capacity is adequate to remove core decay heat and to prevent the pressurizer from becoming solid for the postulated FWLB at the power uprate conditions. The staff has reviewed the assumptions and the results of the licensee's analysis, and finds that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.6.2.15 Inadvertent Loading of a Fuel Assembly into the Improper Position

The licensee analyzed this event using the approved computer code ROCS for calculations of rod worth and power peaking factors.

The fuel misloading events were assumed to initiate from full power for cases with the following conditions: 1) interchange of an exposed assembly with another exposed assembly, 2) interchange of an exposed assembly with a fresh assembly, and 3) interchange of fresh assemblies with different enrichment and/or burnable absorber characteristics. The licensee developed a method for determining if an assembly misloading is detectable based on the tests used to identify assembly misloading during startups. Representative (worst case) assembly misloadings (interchange of two assemblies and mis-orientation of one assembly) were evaluated for the ANO-2 power uprate using the ROCS model for the Cycle 16 core design. The analytical results have met the SRP 15.4.7 acceptance criteria that allow failures of a small fraction of fuel rods in the core for a fuel assembly misloading event. Therefore, the analysis is acceptable.

3.6.2.16 Steam Generator Tube Rupture (SGTR) With or Without a Concurrent Loss of Alternating Current Power

The licensee has analyzed the SGTR accident at the power uprate conditions. The results of its analyses show that the SGTR with a concurrent loss of AC power is limiting. In this limiting case analysis, there is sufficient margin for the operators to take control of the auxiliary feedwater flow rate prior to overfilling the SGs. This is the result of the large secondary volume in the SGs. The licensee stated in its supplemental letter dated October 31, 2001, that the ANO-2 operators are trained to cool and isolate the ruptured SG within 30 minutes. The initiating time is generally considered to begin when the event is diagnosed following completion of standard post trip actions, which typically require 10 minutes to 15 minutes to complete. This will result in a total time of less than 60 minutes for cooldown and isolating the ruptured SG. The licensee has provided the results of its SGTR analyses to isolate and cooldown the SG within 60 minutes following the event. The staff noticed that the licensee did not assume a limiting single failure in the analysis of this accident, which would result in non-conservative consequences. However, this licensee's approach is consistent with its current licensing basis and is, therefore, acceptable. The staff has reviewed the other assumptions and the results of the licensee's analysis, and finds that the other assumptions used in this analysis are conservative and the results of this analysis met the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.6.2.17 Control Element Assembly Ejection

The licensee analyzed the control rod assembly ejection event using the approved STRKIN II computer code for the transient analysis simulation. The input parameters from Tables 7.3.14-1 and 7.3.14-2 of the application, along with physics data and associated uncertainties, were assumed in the analysis, which were consistent with the power uprate and the current analysis in the SAR. Two sets of cases were analyzed by the licensee for this event: one initiated at HFP and one initiated at HZP. The results show that the peak pellet enthalpy remains below the limit of 200 calories per gram (cal/g). For both limiting cases, the ejected three-dimensional (3D) peaks versus ejected worths were generated based on the acceptance criteria for total average enthalpy and incipient centerline melting threshold. No incipient centerline melting fuel failures will occur as long as the cycle-specific data remains within these limits. Up to 15% of the fuel may exceed the acceptance criteria for total average enthalpy. As discussed in Section 7.3.3 of this safety evaluation, the doses associated with up to 15% fuel damage were found to be less than the acceptance criteria noted in SRP 15.4.8 and, therefore, acceptable.

As a result of a fuel failure during a test at the CABRI reactor in France in 1993, and one in 1994 at the NSRR test reactor in Japan, the NRC recognized that high burnup fuel cladding might fail during a reactivity insertion accident (RIA), such as a CEA Ejection event, at lower enthalpies than the limits currently specified in Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." However, generic analyses performed by all of the reactor vendors have indicated that the fuel enthalpy during RIAs will be much lower than the RG 1.77 limits, based on their 3D neutronics calculations. For high burnup fuel which has been burned so long that it no longer contains significant reactivity, the fuel enthalpies calculated using the 3D models are expected to be much lower than 100 cal/g.

The staff has concluded that although the RG 1.77 limits may not be conservative for cladding failure, the analyses performed by the vendors, which have been confirmed by NRC-sponsored calculations, provide reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to 60 gigawatt days per metric ton uranium, such as ANO-2, will neither (1) result in damage to the reactor coolant pressure boundary (RCPB), nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel (RPV) internals to impair significantly the capability to cool the core as specified in current regulatory requirements.

3.6.2.18 Instantaneous Closure of a Single Main Steam Isolation Valve (MSIV)

The instantaneous closure of a single MSIV event was analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The methodology, assumptions, and results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.19 Core Protection Calculator Dynamic Filters Analysis

The CPC dynamic filters were analyzed as a part of the RSG project at an uprated power level of 3026 MWt for Amendment 222. There are no other power uprate changes that impact this event. The results of this analysis were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that this analysis remains valid for supporting the power uprate application.

3.6.2.20 Accident Analysis Evaluation Summary

For those events that were analyzed as a part of the RSG project at an uprated power level of 3206 MWt, the methodology, assumptions and results of those analyses were reviewed and approved by the staff in its safety evaluation for Amendment 222. The staff finds that those analyses remain valid for supporting the power uprate application.

For those events that were not analyzed as a part of the RSG project in Amendment 222, the staff has reviewed the assumptions and the results of the licensee’s new analyses and finds that the assumptions used are conservative and the results meet the acceptance criteria for those events. Therefore, the staff finds the licensee’s analyses acceptable.

3.7 Anticipated Transients Without Scram (ATWS)

The licensee has implemented the ATWS rule, 10 CFR 50.62, at ANO-2 by installing a diverse scram system (DSS), diverse turbine trip (DTT), and diverse emergency feedwater actuation system (DEFAS). These system designs were approved by the NRC based on their reliability, independence, and diversity from the plant protection system.

The licensee has evaluated the setpoints for these systems and concluded that they provide adequate protection from an ATWS event. The actuation setpoint for DSS/DTT remains above the reactor protection system (RPS) high pressurizer pressure setpoint and below the pressurizer safety valve lifting pressure. The actuation setpoint for DEFAS is below the plant protection system setpoint for the emergency feedwater actuation system. The staff has reviewed the licensee’s evaluation and finds that the design at ANO-2 regarding ATWS remains effective for power uprate and is therefore, acceptable.

3.8 Station Blackout

Station blackout is defined in 10 CFR 50.2 as the complete loss of preferred offsite and Class 1E onsite emergency AC power systems. The licensee has performed an evaluation of the station blackout analysis to determine the impact of power uprate. The licensee stated that there was no coping analysis performed or required for ANO-2 because an alternate AC (AAC) diesel generator (DG) has been installed and the AAC DG has been demonstrated, by testing, to be available to power the shutdown buses within 10 minutes of the onset of station blackout. The AAC DG loads during accident conditions will continue to be within the design and licensed ratings of the machine, which has a continuous duty rating of 4400 kilowatts at 4.16 kilovolts (kV) (0.8 power factor). The AAC generator is sized to carry the largest load from either of the two ANO, Unit No. 1 (ANO-1) safety buses or either of the two ANO-2 safety buses. The current accident loading calculation is conservative since the modification to the containment cooler fans actually results in a load reduction. The staff finds the AAC power source adequate for power uprate.

3.9 Nuclear Fuel

3.9.1 Thermal-Hydraulic Analyses

Cycle 16 uses the same fuel design as that used in Cycle 15 with erbia replacing gadolinia as the burnable poison. As such, the licensee analyzed the impact of the power uprate on the thermal-hydraulic parameters affecting the DNBR. Steady state analyses were performed of the bounding cycle design at the uprated power level of 3026 MWt using the NRC-approved computer codes TORC and CETOP.

The effects of fuel bowing as a function of assembly burnup were included in the DNBR calculation. Any bowing penalties were included in the DNBR calculation as per the methodology described in CEN-289(a)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One - Unit 2," dated December 1984.

The thermal-hydraulic analyses for Cycle 16 utilized typical values for the mechanical and thermal-hydraulics data. The values of these data, such as enrichment, MTC, maximum burnup, etc., were taken to be the same as the current licensed limits.

No changes were made to the nuclear design philosophy, methods and models to accommodate the power uprate. The approved methods and models that were used for this power uprate are the same as those used for a typical reload.

The staff finds that the thermal-hydraulic analyses are acceptable because they meet the thermal-hydraulic design criteria, and because they were performed using NRC-approved methodologies, as referenced in Section 8.1.3 of the December 19, 2000, application.

3.9.2 Corrosion Performance and Design Analyses

Recently, increasing cladding corrosion and oxide spalling have been observed in high duty fuel cycles at some Combustion Engineering-designed plants. In a response to a staff concern of cladding corrosion, the licensee stated that a corrosion analysis was performed with fuel management guidelines and limits, including a 100 micron corrosion limit. The results showed that the guidelines and limits are met. The staff reviewed the analysis and finds that the analysis is conservative and thus, acceptable for the ANO-2 power uprate.

The staff finds that the fuel performance analyses are acceptable because they meet the fuel design criteria as referenced in CEN-161(B)-P, Supplement 1-P-A, "Improvement to Fuel Evaluation Model," dated January 1992, and because they were performed using NRC-approved methodologies.

The licensee also provided fuel design analyses including stress, strain, fatigue, rod internal pressure, clad collapse, shoulder gap, and holddown margin. The results showed that the all fuel design requirements and limits are met up to an approved rod average burnup of 60,000 megawatt days per metric ton of uranium (MWD/MTU). The licensee indicated that the results are based on a conservative assumption that the fuel will perform well beyond the current approved burnup limit. The staff reviewed the analyses and finds that these analyses are conservative and thus, acceptable for ANO-2.

Based on the acceptable corrosion performance and conservative fuel design analyses, the staff finds that the fuel design is acceptable for the ANO-2 power uprate up to a rod average burnup of 60,000 MWD/MTU.

4.0 SYSTEMS, STRUCTURES AND COMPONENTS EVALUATION

4.1 Structural and Pressure Boundary Integrity of the Nuclear Steam Supply System and the Balance-of-Plant (BOP) Systems

4.1.1 Reactor Vessel

The proposed power uprate of ANO-2 will increase the operating core power level by approximately 7.5% above the currently licensed level of 2815 MWt. The licensee reported that the power increase will result in the revised design parameters given in Table 3-1 of Enclosure 5 of the application for the proposed power uprate conditions. The structural analysis of the RCS due to deadweight and thermal expansion loads is addressed in Section 5.4 of the application. As part of the RCS system, components needed for maintaining the RCPB consist of the RV, RCPs, SGs, hot- and cold-leg piping, pressurizer, surge line, and supports. The seismic evaluation is described in Section 5.5 of the application using 1% damping for the operating basis earthquake (OBE) and 2% damping for the design-basis earthquake. The dynamic analysis of the RCS with the RSGs is addressed in Section 5.7 of the application for the LOCA conditions.

In its August 23, 2001, response to a staff request for additional information (RAI), the licensee provided the evaluation of the RV for the RSGs and the proposed power uprate. The analyses include the effects of increased deadweight loads due to the SG replacement, the changes in the design-basis thermal hydraulic parameters in Table 3-1 of Enclosure 5 of the December 19, 2000, application, the seismic loads, and the LOCA loads. The calculated stresses and fatigue cumulative usage factors (CUFs) for the most limiting vessel components, as identified in the RV stress reports, are summarized in Attachment 2 of the August 23, 2001, RAI response. The Code and Code Edition used for the RV evaluation is ASME Code, Section III, 1968 Edition, with addenda through the summer of 1970, which is the design code of record. The staff found the licensee's evaluation consistent with the AOR for the ANO-2 RV since the calculated stresses and CUFs are less than the allowable code limits. This is acceptable to the staff.

On the basis of its review, the staff finds that the RV components will be in compliance with licensing-basis codes and standards for the proposed power uprate condition.

4.1.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the effect of the power uprate on the RV internals (RVIs) in Section 5.2 of the application. In the evaluation, the licensee reviewed the effects of SG replacement, power uprate (revised thermal hydraulic parameters), and revised pipe break input data on the existing analysis for the normal operating plus upset condition (Level A+B) and faulted conditions (Level D) documented in the existing AOR for the ANO-2 RVIs. The licensee's structural analysis included the current fuel assembly design and mass, and modeling refinements to the seismic analysis, as well as pipe break analyses due to the use of leak-before-break (LBB) methodology.

The Level A+B and Level D evaluations were performed in accordance with the criteria of Section III, Subsection NG, 1973 draft and Appendix F of the ASME Code, respectively, as defined in ANO-2 SAR Section 4.2.2. In the evaluation of RVI components where the revised data was encompassed by the AOR, the stresses calculated in the AOR were retained as-is, and where the revised data is more limiting than that used in the AOR, the stresses were recalculated using the AOR methodology and revised data. The AOR methodology was also enhanced to address data not previously available. The enhancement involved thermal loadings.

The results of the licensee's evaluation are summarized in Table 5-1 of the application in the form of structural margins. Based on the positive structural margins shown in Table 5-1, the licensee concluded that the stress intensities in the RVI components resulting from the revised loadings associated with the RSGs and the power uprate, satisfy the design criteria for both normal operating-plus-upset and faulted design conditions.

The impact of the revised hydraulic data on the hold-down ring's capability to provide adequate core barrel and upper guide structure hold-down force was also evaluated. By supplemental letter dated August 23, 2001, the licensee provided additional information in its response to a staff RAI pertaining to the hold-down ring capacity. The licensee indicated that the allowable hold-down ring rocking and sliding margins are engineering judgements based on operating experience with numerous plants. The separate allowable margins determined for normal operating and transient conditions provide a sufficiently conservative hold-down force to prevent rocking and sliding of the core support barrel and upper guide structure assemblies. In addition, the calculation of the hydraulic loads used in the evaluation is based on conservative assumptions. The evaluation shows that the hold-down force applied by the ring meets the established allowable margins.

In its August 23, 2001, response to a staff RAI, the licensee also indicated that flow-induced vibration is caused by the application of dynamic hydraulic loads. These dynamic loads are associated with pump pulsation and random turbulence. Hydraulic load input to the RVI structural evaluation included both static and dynamic loads. The resulting RVI structural margins summarized in Table 5-1 of the application, therefore, reflect the application of flow-induced vibration loads. These flow-induced vibration loads also affect high-cycle (i.e, $> 10^6$ cycles) fatigue usage. At the time of the AOR, high-cycle fatigue curves were not included in the ASME Code and high-cycle fatigue evaluations were not, and are not, required by the ANO-2 SAR. For the RSG and power uprate, high-cycle fatigue of RVI components was addressed via a scoping evaluation, which used the high-cycle fatigue curves from current ASME Code Editions to demonstrate that high-cycle fatigue usage would be generally acceptable for ANO-2 RVI components. Low-cycle (i.e., $< 10^6$ cycles) fatigue was addressed by demonstrating that the peak alternating stress required to achieve maximum allowable fatigue usage was greater than that calculated for any of the RVI components.

The staff has completed its review of the RAI response and the licensee's evaluation of the effect of the power uprate on the RVI structural analysis and integrity. The staff concludes that the licensee's evaluation has adequately demonstrated that the RVI will safely withstand the normal operating plus upset and faulted conditions that include the effects of the SG replacement, in addition to the proposed power uprate.

4.1.3 Control Element Drive Mechanisms (CEDMs)

The pressure boundary portion of the CEDMs are those exposed to the vessel/core inlet fluid. The licensee evaluated the effects of the SG replacement and power uprate on the CEDM structural integrity under normal, upset, emergency, and faulted conditions. The evaluation was performed according to ASME Code Section III, 1971 Edition, up to and including the 1973 summer Addenda, which is the Code of record.

In its August 23, 2001, RAI response, the licensee provided information (proprietary) which shows that, for the Level A+B, Level C (emergency conditions), and Level D evaluations, the licensee performed reanalyses where necessary. The licensee concluded that the existing design-basis analyses for the ANO-2 CEDM components' stresses and CUFs remain valid for the proposed power uprate condition. The staff finds the licensee's conclusion acceptable.

On the basis of its review of the licensee's analysis and the associated codes, the staff accepts the licensee's conclusion that the current design of CEDMs continues to be in compliance with the licensing basis codes and standards for the proposed power uprate condition.

4.1.4 Replacement Steam Generators

As part of the SG replacement effort during refueling outage 2R14, the licensee had determined, on the basis of an economic decision, to increase the reactor core power by 7.5% above the current power level of 2815 MWt. The RSG design includes increased tubing surface area to accommodate the power uprate. The tubing surface area in each RSG increases such that it permits a 7.5% power uprate while maintaining the plant thermal hydraulic parameters as shown in Table 3-1 of the application. The system parameters for the condensate, feedwater, and steam supply systems are shown in Table 2-4 of the application for plant operating Cycle 14 (previous operation with degraded OSGs), Cycle 15 (the current operation with RSGs), and Cycle 16 (the proposed power uprate). The licensee indicated that structural integrity analyses were performed accounting for effects of the RSGs and the 7.5% power uprate.

The licensee indicated that the Code and Code Edition used in the evaluation of RSGs for the power uprate is the 1989 Edition of the ASME Code, Section III, which is the Code of record. Both the primary and secondary pressure boundaries (including tubes) of the SGs are designed in compliance with requirements of the ASME Code, Section III, for Class 1 components. Detailed structural analyses have been performed using loading obtained from the applicable SG certified design specification. Table 2-6 of Attachment 2 to the August 23, 2001, supplemental letter summarizes the maximum calculated stresses and CUFs for the tubes, and the primary and secondary pressure boundaries. The staff finds these stresses and CUFs to be less than the Code allowable limits and, therefore, acceptable.

The licensee evaluated the SG supports for the effects of the RSGs and the power uprate. The SG upper supports include the snubber arrangements and the upper Z keys and keyways. The SG lower supports include the sliding base, vertical support pads and anchor bolts, and lower X stop. The licensee indicated that the vertical anchor bolts and the lower X stop will not be affected by the power uprate. The existing snubber supports were evaluated based on the AOR using the 1968 Edition of the ASME Code, up to and including the summer 1970 Addenda. The calculated support loads for the snubber and the Z keys are less than the allowable load limits. Maximum stresses for the lower support sliding base and bolts are provided, respectively, in Tables 2-7 and 2-8 of Attachment 2 to the August 23, 2001, supplemental letter. The Code and Code Edition used for the SG sliding base and bolt evaluation is the 1971 Edition of the ASME Code, up to and including the 1973 Addenda, which is the Code of record. The calculated stresses are less than the Code allowable limits.

The licensee also evaluated the impact of the revised design conditions associated with the 7.5% power uprate on the potential for flow-induced vibration and on the design-basis fatigue analysis for the U-bend tubes. The evaluation of flow induced vibration for the proposed power uprate conditions are summarized in Table 3-1 of Attachment 1 of the August 23, 2001, supplemental letter. The maximum fatigue usage factor for the U-bend tubes is below the allowable limit of 1.0, as shown in Table 2-6 of Attachment 2 to the August 23, 2001, supplemental letter. Table 3-1 of the same letter indicates that flow induced vibrations due to either flow turbulence or vortex shedding are very small, with peak stress far less than the endurance limit. With regard to the fluid elastic tube vibration, the maximum calculated stability ratio for the tubes is less than the ANO-2 allowable limit of 0.75. Therefore, the licensee concluded that the proposed power uprate does not increase the potential for flow induced vibration for the RSG tubes above the allowable limit.

On the basis of its review, the staff accepts the licensee's conclusion that the current RSGs at ANO-2 will continue to maintain their structural and pressure boundary integrity and remain in compliance with the Code of record specified in the SAR, and are, therefore, acceptable for the proposed power uprate.

4.1.5 Reactor Coolant Pumps

In its August 23, 2001, response to a staff RAI, the licensee presented the evaluation of the RCP and motor. The licensee reviewed the existing design-basis analyses of the ANO-2 RCPs to determine the impact of the revised design conditions in Table 3-1 of the application. The licensee's review focused on critical locations such as suction nozzle, vanes, discharge nozzle crotch region, driver mount, and pump casing seal closure, where the components have low stress margin. A finite element analysis was performed using RSGs under power uprate conditions to demonstrate that the most critical stress in the pump components is below the Code allowable limit. The Code and Code Edition used for the evaluation of RCP components is the 1965 Edition of the ASME Code, up to and including the winter 1967 Addenda, which is the Code of record.

The effects of the proposed power uprate were assessed by comparing the accelerations under the revised conditions with the allowable limits defined in the RCP specification. Table 2-4 of the August 23, 2001, supplemental letter summarizes the results of such comparisons. As shown in the table, the maximum accelerations due to the OBE, the safe shutdown earthquake (SSE), and the LOCA are less than the specified design-basis limits.

On the basis of its review, the staff concludes that the RCPs, when operating at the proposed power uprate, will remain in compliance with the requirements of the codes and standards under which ANO-2 was originally licensed.

4.1.6 Pressurizer

The licensee evaluated the structural adequacy of the pressurizer and components for operation at the uprated conditions. The evaluation was performed by comparing the key parameters in the current ANO-2 pressurizer stress report against the revised design conditions in Table 3-1 of the application for the proposed power uprate.

The design and operating temperatures of the pressurizer, and therefore, its thermal movements, are not affected by the proposed power uprate. Minor changes in the main coolant loop piping T-hot and cold-leg temperature (T-cold) values due to RSGs at uprated conditions did not change the thermal anchor movements (TAMs) either at the hot-leg/surge line interface, or along the surge line up to the surge line pressurizer nozzle interface. Also, the design temperature condition (653 °F) affecting thermal expansion of the pressurizer enveloped the current conditions. Since the design-basis transient conditions were not affected, the original thermal analyses remained bounding. Therefore, the original design-basis pressurizer loads and motions remained valid for RSG with power uprate.

In its RAI response dated August 23, 2001, the licensee presented the evaluation of the surge nozzle at the power uprate condition due to branch pipe breaks, which were not previously considered for the surge nozzle. The calculated stresses were below the allowable limits. The Code and Code Edition used for the evaluation is the 1968 Edition of the ASME Code, up to and including the summer 1970 Addenda, which is the Code of record.

On the basis of the above review, the staff concludes that the existing pressurizer components, when operating at the proposed power uprate, will remain in compliance with the requirements of the codes and standards under which ANO-2 was originally licensed. This is acceptable to the staff.

4.1.7 Nuclear Steam Supply System Piping and Pipe Supports

The licensee evaluated the NSSS piping and supports by reviewing the design-basis analysis against the uprated power condition, with regards to the design system parameters, transients, and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop, the RCS tributary lines, and the pressurizer surge line piping systems. The methods, criteria, and requirements used in the existing design-basis analysis, as specified in the ANO-2 SAR, are applicable for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate. The T-hot for the power uprate is projected to be slightly (5 °F) greater than the T-hot at the current rated power level. The T-cold for the power uprate condition will also increase slightly (2 °F) from the current power level. The licensee indicated in Section 5.4.4 of the application that there is a slight increase in thermal loads in the RCS piping due to the power uprate effects. The licensee’s evaluation also indicated that the original design-basis thermal displacement and anchor movement at RCS nozzles are bounding for the power uprate. The seismic response of the RCS piping due to the SG replacement is addressed in Section 5.5 of the application. The licensee confirmed that the original RCS seismic loads, including interface loads, nozzle loads, and component loads remain valid for the power uprate.

The licensee evaluated the LOCA loads on the RCS with RSGs in Section 5.7 of the application. The dynamic analyses were performed to calculate the RCS response to the selected branch line pipe breaks (BLPBs). In Section 5.6 of the application, the licensee discussed the effects of the RSGs and power uprate on the existing LBB analysis. The licensee concluded that all piping loads due to thermal, deadweight, and seismic loads for the power uprate were found less than the envelope loads employed in the CEN-367-A LBB evaluation and are, therefore, acceptable for ANO-2. With application of LBB, the dynamic effects of postulated main loop pipe breaks, which were required and accounted for in the original RCS design, are excluded from the design basis for the RCS piping and components.

Following the application of LBB at ANO-2, the limiting pipe breaks considered in the RCS structural integrity analyses are five BLPBs in the following largest tributary pipes: main steam line, feedwater line, surge line, safety injection line, and shutdown cooling line. As a result of LBB, the current licensing basis LOCA loads are substantially reduced, compared to the original design LOCA loads that consider the postulated main loop pipe breaks. Therefore, the original design-basis loads for the main loop piping, including LOCA loads, remains acceptable for the proposed power uprate.

As discussed in its original application and in response to the staff's questions, the licensee evaluated the NSSS design transients that are used for the structural design of the RSGs, RCS coolant piping, and RCS components. The licensee indicated that the original design transients used in the evaluation of the RCS piping systems and equipment nozzles are in general conservative such that: 1) the number of occurrences assumed exceeds the expected number, 2) conservative methods of predicting the range of pressure and temperature (P-T) for the transients were used, and 3) a composite transient was defined with the most severe portion of the transient derived from a group of transients. As a result of its review, the licensee concluded that the existing stresses, fatigue usage factors, and loads remain bounding for the power uprate for the NSSS components, including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, the pipe supports, and the auxiliary equipment (i.e., heat exchangers, pumps, valves, and tanks). The RSGs were evaluated in an earlier section of this safety evaluation.

The licensee evaluated the surge line considering the thermal stratification transients. The current RCS stratification transients were developed from actual plant operating data, and are a compilation of data from a number of plants. Also, since the temperatures during heat-up and cool-down were not affected by the power uprate, the previous design-basis transient results for the surge line remain unchanged. The licensee also indicated that: 1) the original TAMs at the surge line nozzles did not change due to the power uprate, 2) the surge line nozzle deadweight and thermal loads from the original design basis remain unchanged, and 3) the original response spectrum input for the surge line seismic analysis was found bounding for the RSGs with the power uprate. The licensee also evaluated the response of the surge line to pressurizer motion caused by smaller pipe breaks at the top of the pressurizer, and found that the effects on the surge line were enveloped by the effects of the five major BLPBs. The calculated stresses in the pressurizer surge line nozzle provided in Attachment 2 of the August 23, 2001, supplemental letter are well below the allowable limits. The Code and Code Edition used for the pressurizer surge line nozzle is the 1968 Edition of the ASME Code, up to and including the summer 1970 Addenda, which is the Code of record.

The RCS tributary lines were analyzed to address the changes resulting from the RSGs and the power uprate. The SG replacement necessitated a reconciliation analysis because the RSGs weigh more than the OSGs. When these analyses were performed for the RSGs, the power uprate conditions were included. The licensee indicated that the impact of power uprate itself on the analyses was insignificant. For the main steam and feedwater lines (Class 2 and 3 piping) inside containment, the analyses were performed in compliance with the requirements of the ASME Code, Section III, Subsections NC and ND, 1971 Edition, up to and including the summer 1971 Addenda, which is the Code of record. For Class 1 piping in the safety injection, shutdown cooling, and pressurizer spray lines, the Code and Code Edition used for the evaluation was the 1980 ASME Code, Section III, NB-3600, which is different from the Code of record, the ASME Code 1971 Edition through the summer 1972 Addenda. The licensee indicated that the 1980 Edition, which is a more recent Code than the Code of record, was used in the reanalysis as permitted by the ASME Code. As such, the stress intensity indices in stress equations for various weld and joint types can be more accurately defined. Also, stress allowable limits from the Code of record were used because the materials were certified to meet the Code of record. The staff concludes that the licensee has adequately demonstrated the acceptability of its use of the 1980 Edition to meet requirements of NCA-1140, “Use of Code Editions, Addenda, and Cases,” for using a Code other than the Code of record. Tables 8-1 to 8-6 of the August 23, 2001, RAI response provide calculated maximum stresses, fatigue usage factors, and Code allowable limits for the Class 1 piping of the safety injection, shutdown cooling, and pressurizer spray lines. Tables 8-7 and 8-8 provide calculated maximum stresses and Code allowable limits for the main steam and feedwater piping, which are Class 2 piping and do not require a fatigue analysis. The staff finds the maximum calculated stresses and CUFs to be less than the Code allowable limits and are, therefore, acceptable.

On the basis of its review of the licensee’s submittal, the staff accepts the licensee’s conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the tributary lines connecting to the main loop piping will remain in compliance with the requirements of the design-bases criteria, as defined in the ANO-2 SAR, and are therefore, acceptable for the proposed power uprate.

4.1.8 Balance-of-Plant Piping

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design-basis parameters with the proposed power uprate conditions. The BOP piping systems that are affected by, and evaluated for, the power uprate are listed in Section 2.1 of the application. In its response to the staff’s RAIs, the licensee provided further information in supplemental letters dated August 23 and November 17, 2001.

The licensee evaluated the affected systems on the basis of the uprated input parameters in Table 3-1 of the application for RCS temperatures, and steam temperature and flow rate, considering the heat balance at a reactor core power level of 3026 MWt. Revised pressures and temperatures for the power uprate were reviewed against the existing design-basis calculations for piping stress, pipe supports, nozzles, and anchors. In the majority of cases, the changes in maximum pressure and temperature values due to power uprate were bounded by design pressures and temperatures and, therefore, the original evaluations remained valid. For those systems where the design parameters of the piping were exceeded, a scaling factor was used to increase the stress or calculated load in the AOR by the ratio of the parameter increase. A detailed calculation using the scaling factor was provided by the licensee in the licensee's November 17, 2001, supplemental letter. The calculation included thermal expansion stresses, flow-accelerated corrosion (FAC), piping thermal movement limits in fire barrier penetrations, past flaw evaluations, expansion joints, and dynamic loadings due to fast valve closure transients. The licensee indicated in its August 23, 2001, supplemental letter that the most critical BOP piping systems affected by the power uprate are the main steam and main feedwater piping. The maximum calculated stresses for this piping, considering the proposed power uprate, are less than the Code allowable limits, as shown in Tables 8-7 and 8-8 of the August 23, 2001, supplemental letter.

In addition, the licensee indicated that an improvement was made for the power uprate to consider the turbine stop valve fast closure dynamic loading that was not included in the original design basis for the main steam piping between the stop/control valves and the high pressure turbine inlets (originally supplied by GE as part of the main turbine-generator). As a result, new dynamic forcing functions for power uprate conditions were specifically calculated for this piping. The new dynamic loads on the piping between the stop/control valves and the high pressure turbine inlets, including the steam line nozzle loads on the high pressure turbine casing, were evaluated and found to remain below the design-basis limits.

The licensee evaluated pipe supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for the affected limiting piping systems. The increase in pipe support loads due to the power uprate conditions is consistent with the increase in piping stresses. However, when combining these increases with the loads such as seismic and deadweight, that are not affected by the power uprate, the overall combined support load increases are generally insignificant. The licensee indicated that as a result of the evaluation, there are very few and minor modifications to the piping supports due to the power uprate. For instance, to reduce the nozzle loads on one of the feedwater pump drive turbines, two spring supports were adjusted to new load settings.

In its August 23, 2001, response to a staff RAI, the licensee indicated that the main steam piping has been the system that has displayed the most sensitivity to flow-induced vibration. Because of this history, the licensee, along with a second-party review by Southwest Research Associates, studied the potential changes in the main steam piping vibration due to changes in the pressure and mass flow rate for original design (Cycle 14 - the last cycle with the OSGs), and the first cycle for power uprate (Cycle 16). This study indicated that the kinetic energy is the driving force behind flow-induced vibration of the piping. A summary of the comparison between the original (Cycle 14) and post-power uprate conditions (Cycle 16) is provided in Table 11-1 of the August 23, 2001, supplemental letter. From this table, it can be seen that, although the mass flow rate is increasing, steam velocity and kinetic energy levels will be lower after power uprate than during Cycle 14. In addition, information provided in the licensee's November 17, 2001, supplemental letter demonstrates that the measured vibration stress levels during Cycle 14 and Cycle 15 are less than the endurance limit established by the ASME OM-3 Code, "Operation and Maintenance Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems." The licensee determined it is reasonable to predict that the power uprate operating conditions will not cause unacceptable increases in the vibration of the major main steam piping. The licensee also indicated that the start-up testing program included the installation of vibration monitoring instrumentation on the main steam piping inside containment, hand-held collection of vibration data on main steam piping outside containment, and structured walkdowns of virtually all of the piping outside containment for visual identification of piping vibration. For piping identified by visual walk-downs, vibration data collection was performed.

By letter dated February 7, 2002, the licensee modified its startup testing program to utilize hand-held monitoring devices for collection of vibration data on the main steam piping inside containment rather than rely on installed vibration monitoring instrumentation. The licensee provided the bases for the use of hand-held monitoring devices inside the containment as follows: 1) hand-held vibration monitoring devices provide a measurement accuracy equivalent to the temporary installed instruments which were used during startup following refueling outages 2R14 and 2R15, 2) elimination of installing and removing temporary instruments inside the containment reduces radiation exposure to personnel, 3) use of hand-held instruments allows a visual confirmation of vibration levels in branch piping in addition to the main steam piping, 4) malfunctions of hand-held monitoring instruments can be more easily corrected than temporarily installed instruments, and 5) use of hand-held instruments enables the collection of more data points that are immediately available during startup testing. Based on its review of the licensee’s evaluation discussed above, the staff concluded that the main steam piping is acceptable for the proposed power uprate with regard to the potential flow-induced vibrations, considering the fact that the licensee has implemented a piping walk-down and vibration testing program following the recommendations in ASME OM-3.

As a result of the above evaluation, the staff concludes that the BOP piping remains acceptable and continues to satisfy the design-basis requirements for the power uprate.

4.1.9 Pumps and Valves

As discussed in its original request and response to staff questions, the licensee evaluated the effect of the power uprate on the capability of plant mechanical systems, including safety-related pumps and valves, to perform their safety functions at ANO-2. For example, the power uprate did not increase the minimum performance requirements for safety-related pumps at ANO-2. In evaluating the safety-related air-operated valves (AOVs) for the expected uprate conditions, the licensee found the MSIVs to be the only safety-related AOVs in a system or application impacted by the power uprate, and determined those AOVs to be capable of performing their intended function under power uprate conditions.

The licensee also evaluated the safety-related motor-operated valves (MOVs) within the scope of the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The licensee reviewed fluid temperatures and pressures resulting from the power uprate, and plant calculations for MOV setpoints, expected differential pressures, seismic capability, and structural weak links. The licensee evaluated potential pressure locking and thermal binding of its safety-related power-operated gate valves in light of the proposed power uprate. The licensee determined that the power uprate conditions did not affect the scope of valves evaluated in response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The licensee also determined that the valves previously evaluated in response to GL 95-07 would not be adversely affected by potential pressure locking or thermal binding as a result of the proposed power uprate.

The staff finds the licensee’s evaluation of the effect of the proposed power uprate on the capability of safety-related pumps and valves at ANO-2 to be acceptable, based on the staff’s review of the information submitted by the licensee describing the scope, extent, and results of the evaluation of safety-related pumps and valves at ANO-2.

In regard to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," the licensee indicated that for water-filled piping, the required relief valves have already been installed. The relief valve setpoint will not be affected by the proposed power uprate. The licensee concluded that the current ANO-2 responses to GL 96-06 will remain unchanged for the power uprate. The staff concurs with the licensee’s conclusion based on the fact that the potential for overpressurization of the water-filled piping does not increase due to the proposed power uprate.

4.2 Steam Generator Tube Integrity

The NRC staff reviewed the integrity of SG tubing under the power uprate conditions, specifically on tube wear degradation at anti-vibration bar locations, potential tube degradation other than tube wear, and the 40% tube plugging limit in the plant TSs. The licensee stated that the RSGs were designed and analyzed for power uprate conditions, including consideration of tube degradation and the tube plugging limit. These issues are discussed below.

Tube wear degradation is caused by relatively large tube vibration, resulting in intermittent sliding contact between tube and anti-vibration bars. The primary cause of tube vibration is the hydrodynamic excitation of the tubes by the secondary side fluid in the SGs. Fluid-elastic tube vibration is a concern because it is a self-excited mechanism which may lead to tube instability. Westinghouse has performed testing and evaluated field experience from previous designs to develop analysis methods to assure a significant margin to instability is maintained. Tube support spacing in the anti-vibration bars in the U-bend region is designed so that tube vibration response frequencies will not exceed the instability threshold under the power uprate conditions.

The RSGs include several features that would minimize the potential for tube wear at the anti-vibration bar locations. These features include use of stainless steel anti-vibration bars, control of anti-vibration bar thickness, and control of tube ovality to assure tight tolerances. The five sets of anti-vibration bars in the U-bend region provide redundancy so that all the tubes remain fluidelastically stable, even if it is assumed that some of the support points are inactive. The anti-vibration bars in adjacent columns are inserted to different depths to minimize the formation of flow stagnation regions with resulting deposition of sludge. The supports are oriented to minimize contact length and potential for crevice corrosion.

The licensee concluded that analyses and tests demonstrate that unacceptable tube degradation from the anti-vibration bars will not be expected when operated at power uprate conditions. Operating experience with SGs having the same size tubes and similar flow conditions supports this conclusion. Based on the above, the NRC staff finds the licensee’s conclusion acceptable.

With respect to potential tube degradation (other than tube wear), the RSG design has several enhancements to minimize tube degradation. The tubing is made of thermally treated Alloy 690 material. Testing and field experience have shown it to be more resistant to corrosion cracking than the mill-annealed Alloy 600 tubing in the OSGs. The tube-to-tubesheet joints have been installed with closely controlled hydraulic expansions to minimize cracking at the joints. The tube support plates are made of stainless steel material which resists corrosion better than the carbon steel material. The tube support plates are designed with the broached configuration and shortened contact length to minimize stagnation regions. The staff finds that the enhancements in the RSGs will minimize potential tube degradation under the power uprate conditions.

The licensee analyzed the 40% tube plugging limit and the results are shown in Topical Report WCAP-15406, “Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators,” which was submitted to the NRC on July 19, 2000. The analysis determined that the limiting condition for establishing the plugging limit is maintaining the required margin to tube burst for the normal operating pressure differential. The analysis assumed the bounding case of reduced secondary pressure to account for future tube plugging. For the case of the unplugged condition, power uprate steam pressure will be increased, normal operating pressure differential is therefore reduced, and structural margins are increased. The licensee concluded that the 40% tube plugging limit in the plant TSs is not affected under the power uprate conditions. Based on the above, the staff finds the licensee's conclusion acceptable.

4.3 Primary and Secondary Water Chemistry

The licensee has evaluated the effects of power uprate on the primary and secondary water chemistry. The evaluation concluded that these programs are suitable for power uprate conditions of flow, pressure, and temperature. In addition, the effects of the increase in primary and secondary water volumes were included in the evaluations. The primary and secondary sampling systems are suitable for the expected small increase in primary temperatures (approximately 2 °F to 5 °F) due to power uprate.

Components made of copper in the feedwater system had affected the OSGs. As a result, copper components were removed from all fluid systems in the condensate, feedwater, steam supply, extraction, and drain systems, and replacement components in these systems are suitable for power uprate.

The primary-side chemistry incorporates the use of lithium hydroxide for pH and corrosion control and hydrogen (H_2) for oxygen scavenging. However, for Cycle 15, a new lithium strategy was started which maintains pH at 7.1 and allows lithium levels to exceed 3.5 parts per million for less than 17 days. This new strategy, based on a Westinghouse evaluation, can minimize corrosion product release and crud deposition following SG replacement. As a result of implementing this new strategy, SAR Section 9.3.4.2.4 was revised.

The secondary-side chemistry will have higher flow rates due to power uprate. However, the licensee has concluded that the current system is adequate to meet the needed injection rates. In addition, the licensee's chemistry department has obtained industry modeling software to predict chemistry changes based on various parameters including temperature and flow.

Based on the information provided by the licensee, the staff concludes that the proposed power uprate will not cause any detrimental effects to the operation of the primary and secondary chemistry programs.

4.4 Alloy 600 Program

Section 4.1.1.1 of the license amendment application assessed the effect of the increase in power on integrity of the Alloy 600 components in the RCPB. The licensee identified that the following RCPB components are fabricated from Alloy 600:

- resistance temperature detector nozzles to the RCPB piping
- instrumentation nozzles and heater sleeves in the pressurizer
- CEDM nozzles to the RV head
- in-core instrumentation nozzles to the RV head
- vent lines to the RV head

Since power uprate results in only a small increase in T-cold and T-hot (T-cold increases from 549 °F to 551 °F and T-hot increases from 604 °F to 609 °F), the inspection program for the resistance temperature detector nozzles in the RCPB piping will be adequate. In the power uprate application, the licensee stated that the ANO-2 pressurizer will continue to operate at the same temperatures as before, and that therefore the potential for primary water stress corrosion cracking (PWSCC) to occur in the pressurizer instrumentation nozzles and heater sleeves would not increase. In addition, in a supplemental letter dated February 7, 2002, the licensee provided a revised response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which required all holders of operating licenses for pressurized water reactors (PWRs) to provide information related to structural integrity of the reactor pressure vessel head penetration (VHP) nozzles (which include the CEDM, in-core instrumentation, and vent line nozzles). The response was to include the extent of nozzle leakage and cracking that had been found to date, the inspections and repairs that had been undertaken, and the basis for concluding that the licensee's planned inspections would ensure compliance with applicable regulatory requirements.

In response to NRC Bulletin 2001-01, the Materials Reliability Program (MRP) Alloy 600 Issues Task Group produced a report, "PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," dated August 2001. The MRP-48 report included a "susceptibility ranking" parameter of each PWR unit which was defined to be the operating time in effective full power years (EFPY) required for the unit to reach an effective time-at-temperature equivalent to that of Oconee Nuclear Station Unit 3 (ONS-3) in early 2001 (i.e., the time at which circumferential cracks were identified in the base material above the J-groove weld of VHP nozzles at ONS-3). The susceptibility ranking for ANO-2 given in the MRP-48 report was 17.1 EFPYs and was based on the current RPV head temperature. The susceptibility ranking given in the MRP-48 report did not account for changes in the ANO-2 RPV head temperature which will occur in ANO-2 Operating Cycle 16 and subsequent cycles as a result of the ANO-2 7.5% power uprate.

The licensee revised the susceptibility ranking given for ANO-2 in the MRP-48 report by correcting the RPV head temperature to account for power uprate conditions. The licensee's revised analysis characterized the RPV head temperature as a function of T-hot. The value of T-hot for ANO-2 operating cycles through Cycle 15 was 604 °F. The value of T-hot for ANO-2 Operating Cycle 16 and subsequent cycles was taken as 609 °F. Based on the licensee's revised analysis, the susceptibility ranking for ANO-2 was reduced from 17.1 EFPYs to 14.2 EFPYs. Using the criteria in NRC Bulletin 2001-01, the licensee concluded that the ANO-2 reactor pressure VHP nozzles would continue to fall within the "moderate" category (defined as having a susceptibility ranking between 5 and 30 EFPYs) with regard to circumferential VHP PWSCC susceptibility.

The NRC staff performed independent calculations to confirm the susceptibility ranking of the ANO-2 reactor pressure VHP nozzles. The NRC staff confirmed that the ANO-2 reactor pressure VHP nozzles would continue to fall within the "moderate" category.

Since the ANO-2 pressurizer will continue to operate at the same temperature as before the power uprate and the ANO-2 reactor pressure VHP nozzles continue to be ranked in the "moderate" category, the potential for PWSCC to develop in Alloy 600 nozzles will not be significantly affected by the power uprate. Therefore, no change is required for the inspection of Alloy 600 components in the ANO-2 RCPB.

4.5 Chemical and Volume Control System (CVCS)

The primary design function of the CVCS is to maintain RCS inventory and control RCS chemistry. To meet this function, the CVCS must maintain the chemistry of the RCS, maintain RCS inventory, control boron concentration in the RCS, provide auxiliary pressurizer spray, measure RCS leakage, provide for functional testing of safety injection system check valves, measure fission product activity, monitor and collect RCP seal controlled bleedoff, provide capability of testing the RCS pressure boundary, and provide one of several alternate makeup sources to the RCS.

The replacement of SGs will result in a higher RCS inventory. This results in a wider range of RCS temperature changes and, therefore, a greater volume change than before power uprate. However, the licensee has concluded that these changes are within the capability of the CVCS and that no modifications are needed to meet the increased volume requirements.

The power uprate will change the operating temperatures of the RCS, which could have some impact on these CVCS functions. However, the licensee's analysis indicates that the projected RCS letdown temperature for power uprate is very close to the original operating assumption for the CVCS. The original assumed letdown temperature was 550 °F and the letdown temperature under the uprated conditions is expected to be 551 °F.

The licensee has also determined that the ability of CVCS to provide the required shutdown margin is adequate with the exception of the boric acid makeup (BAM) tanks. Section 9.3.4 of the SAR explains that the letdown loop of the CVCS is not required to achieve cold shutdown. In this case of cooldown without letdown, the total makeup added to the RCS is limited by the total RCS shrink during cooldown. Due to the core design for power uprate, the licensee indicated that the minimum required boron concentration in the BAM tank would need to be increased to accommodate this situation. The licensee administratively controls necessary changes to the boron concentrations and operability statements for BAM tanks in Technical Requirements Manual (TRM) Sections 3.1.2.7 and 3.1.2.8, and in TRM Figure 3.1-1. Based on the licensee's evaluation of the CVCS, it will be necessary for the licensee to administratively amend TRM Sections 3.1.2.7 and 3.1.2.8, and TRM Figure 3.1-1, as appropriate, to reflect any necessary changes to the boron concentrations for the BAM tanks. (The licensee provided, for the staff's information, revisions to the TRM sections discussed above that they plan to implement for the power uprate. The licensee does not need NRC approval to revise the TRM.) The administrative changes to the BAM tank boron concentrations will ensure that the CVCS will continue to be capable of maintaining shutdown margins within the limits provided in the COLR, if the shutdown margins fall below those required under the scope of TS 3/4.1, "Reactivity Control Systems."

Based on the information provided by the licensee, the staff concludes that the proposed power uprate and corresponding TRM changes will not impede the operability of the CVCS or the ability of the CVCS to maintain the shutdown margins for the reactor within the limits required in the TSs.

4.6 Leak Before Break Evaluation

The staff has reviewed the information submitted by the licensee regarding the potential impact of the proposed ANO-2 power uprate on the acceptability of the LBB status of the ANO-2 main coolant loop (MCL) piping. The primary system pressure, primary system temperature, material properties, and design-basis SSE loadings are the parameters that could have a significant impact on the facility's LBB evaluation. However, relative to the margins and assumptions in the existing ANO-2 LBB analysis, the requested 7.5% power uprate results in minimal changes to these parameters. Therefore, the staff has concluded that the changes to the LBB evaluation for this piping resulting from the proposed power uprate will not alter the previous conclusions associated with Combustion Engineering Topical Report CEN-367-A demonstrating LBB behavior of the ANO-2 MCL. Therefore, the staff has concluded that, per the provisions of 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-4, the dynamic effects from postulated breaks of the ANO-2 MCL may continue to be excluded from the licensing basis of the facility for post-power uprate conditions.

4.7 Reactor Vessel Fluence and Pressure-Temperature Limits

In the application for power uprate, the licensee did not propose to change the RV P-T curves. The plant TSs indicate that the current P-T curves are applicable to 21 EFPYs of operation. However, the licensee stated that the P-T curves are actually valid to 17 EFPYs. This came about in 1992 in the response to GL 92-01, "Reactor Vessel Structural Integrity," when it was discovered that the critical element (the element with the highest adjusted reference temperature (ART)) was not the intermediate shell plate C8009-1, but the lower shell plate C8010-1, as indicated in the licensee's supplemental letter dated July 24, 2001.

The licensee intends to retain the current P-T limits to the end of Cycle 15, to an estimated 16.96 EFPYs with the 7.5% power increase commencing at the beginning of Cycle 16. This is based on the claim that the fluence value includes a large conservatism and that the new critical element 1/4T and 3/4T ARTs are conservative with respect to the old critical element (where T is the wall thickness). These two issues (fluence evaluation and ARTs for the new critical element) are the subject of this review.

4.7.1 Neutron Fluence

"Summary Report on Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Arkansas Nuclear One Unit 2 Generating Plant," dated May 1, 1984, by L.M. Lowry, et. al., Battelle Columbus Laboratory, summarizes the 97-degree capsule evaluation. That capsule was removed ahead of time (at the end of the second cycle with 1.69 EFPYs of exposure and with the staff's approval) in order for ANO-2 to participate in an extensive cavity dosimetry program carried out by the Electric Power Research Institute (EPRI) and the University of Missouri at Rolla. The cross sections used in the capsule analysis by the Battelle Columbus Laboratory would tend to underestimate the value of the vessel fluence. This condition applies to all plants which used the Evaluated Nuclear Data File/Version B-IV (ENDF/B-IV) or earlier versions of ENDF/B for the transport cross sections for components with high iron content, such as the thermal shield. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001, specifies that the latest ENDF/B library be used, which remedies this shortcoming. However, ANO-2 does not have a thermal shield; therefore, this underestimation is minimal. On the other hand, the neutron source was that corresponding to the first and second cycles, both of which had fresh assemblies in the outer rows. Fresh assemblies maximize the neutron source with respect to the equilibrium cycles (i.e., after the 4th cycle). The licensee used the estimated fluence at 1.69 EFPYs to extrapolate to 21 EFPYs. The current P-T curves are based on this extrapolation, which is conservative. In addition, at the 6th cycle, ANO-2 initiated low leakage core loadings as it transitioned to 18- and 20-month cores. Thus, the fluence projections become more conservative. Entergy did not take credit for the low leakage loadings.

The vessel cavity surveillance project also indicated that the 97-degree projections were conservative by a large margin. The staff attributed a large measure of credibility to these results, because the cavity measurements and surveillance was a research project with many controls and checks. Finally, the preliminary results of the second surveillance capsule, which was removed at the end of the 14th cycle with an estimated exposure of 15.7 EFPYs, also indicated that the estimated exposure for the current P-T curves is conservative. The analytical calculations for the second surveillance capsule were performed using the Framatome Technologies Incorporated fluence methodology (BAW-2241PA), which adheres to the guidance in RG 1.190 and has been approved by the staff.

Based on the above, the staff concludes that the fluence value used in the calculation of the current P-T curves is conservative with a significant margin.

4.7.2 Adjusted Reference Temperatures for the New Critical Element

As indicated above, the ANO-2 response to GL 92-01 changed the vessel beltline critical element from the intermediate shell plate C8009-1 to the lower shell plate C8010-1. The combination of the initial RT_{NDT} and the copper and nickel contents was such that the new critical plate ARTs supported 17 EFPYs of operation with the existing P-T curves. Comparison of Tables 4 and 5 of the July 24, 2001, supplemental letter indicates that the ART values are somewhat conservative for 17 EFPYs. Finally, the staff notes that the licensee states in the same letter, that the new P-T curves for ANO-2 will incorporate surveillance capsule results and include the 7.5% power uprate that is scheduled for Cycle 16 forward, and that the new P-T curves will be submitted to the NRC in sufficient time to allow six months for NRC review and approval.

In summary, the staff finds that the current P-T curves were based on a very conservative fluence and that the new critical element has conservative ART values with respect to the current P-T curves. Thus, the staff finds the present P-T curves acceptable for operation through Cycle 15.

4.7.3 New Pressure-Temperature and Low-Temperature Overpressurization (LTOP) Limits

As indicated above, the licensee did not submit revised pressurized thermal shock (PTS), upper-shelf energy (USE), or P-T limit assessments as based on the uprated neutron fluence values for the RV, in its power uprate application. In a June 7, 2001, RAI, the staff requested that the licensee either: 1) provide technical analyses to demonstrate the P-T, LTOP, PTS, and USE analyses in the current ANO-2 licensing basis will remain valid (are bounding) for the neutron fluences that are estimated to result from the 7.5% increase in rated-power, or 2) provide revised P-T, LTOP, PTS, and USE analyses that are based on the neutron fluences that are estimated to result from the 7.5% increase in rated power, if the fluences used for the current P-T, LTOP, PTS, and USE analyses would not bound those that will result from the 7.5% increase in rated power.

In its August 7, 2001, response to an RAI, the licensee indicated that the validity of the current

P-T limit, LTOP limit, PTS, and USE analyses were demonstrated in response to GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated June 18, 1997. These analyses addressed the change in the limiting RPV beltline material for ANO-2. The licensee has indicated that the fluence values used in the these analyses did not account for the 7.5% increase in reactor power. By supplemental letter dated July 24, 2001, the licensee addressed, in part, the conservatism of the fluence used for these assessments out to 17 EFPYs.

In its August 7, 2001, response to an RAI, the licensee also indicated that new P-T and LTOP limits for ANO-2 will incorporate all surveillance capsule results, including the results from the capsule most recently removed from the ANO-2 RPV, and will include calculated increases to the fluence that result from the 7.5% increase in rated power (i.e., commencing with Cycle 16). The licensee has indicated that Cycle 16 is scheduled to commence on May 18, 2002, and that it will submit the revised P-T and LTOP limits in sufficient time to allow six months for NRC review and approval. The licensee has indicated that it will include new PTS and USE evaluations for the ANO-2 RPV with its submittal of the new P-T and LTOP limits.

The staff considers that P-T and LTOP limits are an integral part of ensuring the structural integrity of the RCPB during normal, transient, and pressure testing operating conditions. To allow the staff ample time to perform its review, and the licensee to implement the new curves once they are approved, the staff requested that the new P-T and LTOP limits for ANO-2 be submitted for review by December 2001. By separate license amendment request dated October 30, 2001, the licensee submitted new P-T and LTOP limits for NRC review and approval. The licensee requested that the new P-T and LTOP limits be approved prior to the end of refueling outage 2R15. This P-T/LTOP limits application was handled separately from the power uprate application, and it was approved by the staff in its safety evaluation for Amendment 242, dated April 15, 2002.

4.8 Environmental Qualification (EQ) of Electrical Equipment

The licensee has reviewed the EQ of electrical equipment located in containment to the revised LOCA profile from the ANO-2 containment uprate and SG replacement efforts, which included power uprate. The equipment remains qualified to the revised LOCA temperature and pressure conditions. The normal and accident radiological conditions for power uprate were also evaluated. The equipment remains qualified to the revised containment radiological conditions. In addition, the revised high energy line break (HELB) temperature and pressure conditions in the auxiliary building were evaluated to determine impact to the current qualification of electrical equipment. The equipment located in the auxiliary building remains qualified to the revised HELB conditions.

Section 9.4.5 of Enclosure 5 of the December 19, 2000, application, discussed non-conservatisms that were discovered in some of the design inputs used in the radiological EQ dose calculations. The non-conservative design inputs were not related to the increase in power and the conclusions related to power uprate were not expected to be affected. Corrective actions were assigned in accordance with the station corrective action program and a commitment was made to notify the NRC staff of the results. In the October 30, 2001, supplemental letter, the licensee informed the NRC that it has corrected the calculations with no impact to the original conclusions regarding power uprate.

The staff reviewed the EQ of electrical equipment results and the non-conservatisms identified by the licensee in the radilogical EQ dose calucations. Based on the staff's review of the licensee's analysis and the criteria set forth in 10 CFR 50.49, the staff finds the EQ of electrical equipment acceptable at the power uprate condition.

4.9 Instrument Setpoints

The proposed amendment reflects instrument setpoint changes consistent with a requested thermal power uprate for ANO-2 from 2815 MWt to 3026 MWt. In response to a staff RAI, the licensee provided further information and clarifications by supplemental letter dated June 26, 2001. The staff's evaluations of the setpoint changes for the identified instrumentation for the new power level are predicated on the assumption that analytical limits used by the licensee are based on the application of approved design codes.

4.9.1 Suitability of Existing Instruments

In its submittal, the licensee stated it has evaluated each existing instrument of the affected NSSS and BOP systems to determine its suitability for the revised operating range of the process parameters affected by the proposed power uprate. Where operation at the uprated power levels impacts the safety analysis limits, the licensee verified that the acceptable safety margin was maintained under all conditions and, where necessary, revised the setpoint and uncertainty calculations for the affected instruments. Apart from the few devices that needed changes, the licensee found that most of the existing instruments are acceptable for operation at the proposed uprated power level. The licensee performed some analyses previously which were reviewed and approved by the staff when the ANO-2 SGs were replaced. The licensee also performed the review of the BOP and NSSS systems, and determined that the instrumentation and control aspects of these systems continue to meet the system requirements and need minor changes. The licensee identified the following changes:

- Setpoints for the ATWS system and the DEFAS may need adjustment to reflect the actual operating conditions for Cycle 16,

- The excore and incore detector circuits will need a minor scaling adjustment to ensure that they read correctly at the uprated power level,

- The plant computer and safety parameter display system (SPDS) software will need minor adjustment (higher power level, steam flow, etc.) for the new operating conditions,

- The feedwater pump speed for Cycle 16 may require adjustment based on the actual operating experience,

- Setpoints for the COLSS and the CPCs will be adjusted based on the power uprate conditions,

- The SG high-level trip setpoint (located in the TRM) may need adjustment based on the power uprate condition, and

- The plant protection system pressurizer pressure low setpoint will be adjusted based on power uprated conditions.

The licensee will make these changes as needed to accommodate the revised process parameters. The staff finds that the licensee's conclusion that the ANO-2 instrumentation and control systems will accommodate the proposed power uprate without compromising safety, once the licensee implements the above noted changes during the next refueling outage, is acceptable.

4.9.2 Reactor Protection System/Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoint and Allowable Values

In its application dated December 19, 2000, the licensee stated that instrument setpoints in the TSs are established using the setpoint methodology which was previously reviewed by the staff and found acceptable for establishing new setpoints in previous amendment applications. However, the staff was concerned about the use of this setpoint methodology for the power uprate application as this methodology was reviewed and approved for specific license amendments. During a conference call on April 26, 2001, the staff requested the licensee to discuss the compliance of their instrument setpoint methodology with respect to Instrument Society of America (ISA) Standard 67.04, "Setpoints for Nuclear Safety-Related Instrumentation," and RG 1.105, "Instrument Setpoints for Safety-Related Systems." In its response dated June 26, 2001, the licensee stated that the methodology meets the intent of both ISA Standard 67.04 and RG 1.105. However, the licensee stated that the methodology does not meet the 95/95 criterion, where the licensee does not have sufficient information to certify that all vendor information provided meets the requirements or where the information is completely missing and they have to use engineering judgement. The licensee further stated that in these cases, care is taken in the calculations to conservatively interpret vendor and field calibration data. In addition, the licensee stated that care is taken to sort random error components from non-random components, and room temperatures are typically based on the worst-case normal and accident conditions to obtain the highest uncertainty, unless less extreme conditions are specifically justified by the calculation. Also, for any safety-related instrument that falls outside the as-found calibration tolerance, root cause evaluation will determine the cause of the abnormal drift in the instrument and the problem will be corrected. The staff finds the licensee's response acceptable.

The proposed setpoint changes associated with the proposed power uprate are intended to maintain existing margins between operating conditions and the reactor trip setpoints, and they do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not adversely affected by the setpoint changes to accommodate the proposed power uprate.

4.9.3 Instrument Setpoints Summary

Based on the above review and justifications, the staff concludes that the licensee's instrument setpoints methodology for the proposed power uprate are consistent with the ANO-2 licensing basis and, therefore, are acceptable.

4.10 Testing

The staff's review of the amendment request focused on the information provided by the licensee in its December 19, 2000, application, Enclosure 5, Section 9.11. In conducting its review, the staff considered the criteria in WCAP-10263. Consistent with the approach used by the licensee in formulating its application, the staff also considered relevant guidance in NEDC-31897P-A.

The licensee submitted its request for NRC review and approval of a 7.5% power uprate license amendment for ANO-2 in accordance with the requirements of 10 CFR 50.90. The staff's review of license amendments is governed by the requirements in 10 CFR 50.92. According to 10 CFR 50.92 requirements, in determining whether an amendment to a license or construction permit will be issued to an applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses or construction permits to the extent applicable and appropriate.

In part, 10 CFR 50.34 requires that an applicant for a license to operate a production or utilization facility include the principal design criteria for the proposed facility in the SAR. Appendix A to 10 CFR Part 50 states that these principal design criteria are to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety, i.e., structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The applicant is also required to include plans for preoperational testing and initial operations in the SAR.

Criterion XI of Appendix B to 10 CFR Part 50 requires that a test program be established to ensure that structures, systems, and components satisfy performance requirements established in accordance with Appendix A to 10 CFR Part 50.

RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," describes the general scope and depth of initial test programs acceptable to the NRC staff to satisfy the applicable requirements in 10 CFR 50.34, and Appendices A and B to 10 CFR Part 50.

In Enclosure 5, Section 9.11, of the December 19, 2000, application, the licensee states that tests in the power ascension test program for operational Cycles 15 and 16 were designed to confirm that the affected systems/components operate within their design and licensing basis, with some tests designed to determine that sufficient margins exist for the planned uprate condition.

ANO-2 TS 6.9.1.1 requires, in part, that a summary report of plant startup and power escalation testing be submitted to the NRC following: 1) receipt of an OL; 2) an amendment to the license involving a planned increase in power level; 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier; or 4) modifications that may have significantly altered the nuclear, thermal, and hydraulic performance of the plant. In its August 30, 2001, response to a June 1, 2001, staff RAI, the licensee stated that during refueling outage 2R14 in the fall of 2000, ANO-2 altered the nuclear, thermal, and hydraulic performance of the unit by installing RSGs. Following replacement of the station's SGs during 2R14, a startup report dated June 11, 2001, was submitted to the NRC that summarized the testing conducted during startup and power escalation to demonstrate satisfactory plant performance with the new SGs. Following refueling outage 2R15 (i.e., the beginning of operational Cycle 16), which is scheduled for the spring of 2002, ANO-2 will uprate power by 7.5%. Following the 2R15 power uprate outage, a similar startup test report will be provided.

Development of the ANO-2 test programs for Cycle 15 and Cycle 16 were based upon the licensee's review of the ANO-2 SAR, the scope of the ANO-2 modifications performed, the scope of testing completed during other SG replacements and power uprate projects, and industry-wide experience from startup testing. The licensee provided a summary of its Cycle 15 and Cycle 16 test programs in its August 30, 2001, response.

Additionally, the licensee identified the following commitments regarding post-power uprate startup testing at ANO-2 in its August 30, 2001, response:

- A similar report (startup test report) will be provided to the NRC following the 2R15 power uprate outage.

- Power will be increased in 2.5% increments from 90% to 100% reactor power. Steady-state operating data will be evaluated against design predictions and any discrepancies will be resolved prior to proceeding with power ascension. The Test Working Group, made up of senior ANO plant management and experienced testing personnel, will review any significant testing deficiencies or anomalies before recommending power ascension to each of the plateaus. Completion of required testing, review and approval of testing results, resolution of testing deficiencies, and approval to proceed in power ascension, will be controlled by a special work plan.

- Projections will be made based upon actual plant data from the current 100% power level of 2815 MWt to the new 100% power level of 3026 MWt. These projections support ANO-2's power uprate design and form the basis for the acceptance criteria at each power plateau level specified (90%, 92.5%, 95%, 97.5%, and 100% of rated power).

- Minor tuning and setpoint design changes are planned for 2R15. These changes will require steady state control system monitoring/testing to be performed at 90%, 92.5%, 95%, 97.5%, and 100% of uprated conditions.

On the basis of the information provided by the licensee, the staff finds that the scope and depth of the ANO-2 post-power uprate startup and power escalation test program: 1) is commensurate with a 7.5% power uprate and corresponding plant modifications, 2) is consistent with the guidance in RG 1.68, and 3) satisfies the applicable requirements in 10 CFR 50.34, and in Appendices A and B to 10 CFR Part 50. Accordingly, the startup and power escalation test program outlined by the licensee will demonstrate that ANO-2 meets performance requirements that satisfy design and licensing basis criteria for the uprated operational envelope and is, therefore, acceptable.

5.0 BALANCE-OF-PLANT SYSTEMS AND RELATED EVALUATIONS

5.1 Background

ANO-1 and ANO-2, are currently operating at their original licensed power levels of 2568 MWt and 2815 MWt, respectively. Until now, the licensee had not requested a license amendment to increase the thermal power for either unit. ANO-1's full power OL was issued on May 21, 1974, and ANO-2's full power OL was issued on September 1, 1978. This amendment proposed changes to the Unit 2 TSs and OL to allow plant operations at an increased reactor core power of 7.5%, from 2815 MWt to 3026 MWt. Prior to requesting this amendment, the licensee made several significant modifications in preparation for the power uprate. In the last refueling outage (2R14), the licensee replaced the existing SGs with SGs which could support the increased power. In addition, the licensee replaced the high pressure turbine steam path components and four stages of the low pressure turbine with newer designs to support operations with the RSGs and to optimize performance at the uprated power. The RSGs are also equipped with integrated steam flow restrictors which limit steam flow during a main steam line break (MSLB). As a result of the containment analyses performed for the replacement of the SGs, several key containment systems and limits needed to be modified. On November 13, 2000, the Commission issued Amendment 225, which approved an increase in the containment building design pressure from 54 pounds per square inch, gauge (psig) to 59 psig. In Amendment 226, also dated November 13, 2000, the TSs for the operation of the containment cooling system (CCS) were revised to require two independent containment cooling groups to be operable with two operational cooling units in each group, in Modes 1, 2, 3, and 4. This TS change was made due to a reduction in the heat removal capacity of an individual cooling unit resulting from the increase in the containment design pressure and a reduction in the pitch of the cooling unit's fan blades.

5.2 Spent Fuel Pool (SFP) Systems

The SFP systems are designed to remove the decay heat released from the spent fuel assemblies stored in the SFP and to maintain purity of the pool's water. Two systems accomplish these functions: the cooling system and the purification system.

5.2.1 Spent Fuel Pool Cooling System

The SFP cooling system consists of two 50% capacity pumps and one 100% capacity heat exchanger. The system is designed to maintain the SFP temperature less than or equal to approximately 150 ^{o}F, assuming the maximum heat load associated with a full core discharge and a service water system (SWS) inlet temperature of less than or equal to 85 ^{o}F. The licensee minimizes the potential of exceeding the SFP design basis by administratively controlling fuel offloads. Prior to an offload, the actual service water temperature and the decay heat load of the fuel to be unloaded are compared to the AOR in the SAR for SFP cooling system heat removal capacity. Offloading could proceed if the actual values for service water temperature and decay heat load were bounded by the AOR, and all other requirements were met.

In its May 30, 2001, response to an RAI regarding the power uprate submittal, the licensee described a change to the methodology used to determine if fuel could be offloaded. An enhancement was made to the AOR by the addition of a graph showing SFP heat exchanger capacity versus service water temperature and pump configuration. The calculated decay heat load will now be compared against the graph to determine the maximum service water temperature allowable for a full core discharge. If the actual service water temperature exceeds the maximum allowable for the heat load, the core offload will be delayed until the service water temperature or the decay heat load decreases sufficiently to be bounded by the graph. The maximum theoretical heat load for a full core offload after the power uprate is 38.10 MBtu/hr. To maintain 150 ^{o}F at this heat load, the maximum service water temperature is 78 ^{o}F. The service water temperature is usually less than 78 ^{o}F, except during the summer months. The licensee concluded that, with the use of administrative controls for controlling fuel offloads, the heat removal capacity of the SFP cooling system is sufficient for the decay heat loads at the uprated power level.

The licensee also evaluated the effect of the proposed power uprate on the SFP emergency cooling mechanism. The current emergency cooling mechanism for the SFP, as described in the SAR, is evaporation and boil-off. Essential service water is the Seismic Category I-assured source of makeup to the SFP to maintain water level. The SFP can be supplied by either loop of service water. The licensee concluded that the heat removal capacity of the service water makeup to the SFP is more than sufficient for the decay heat loads at the uprated power level.

The staff finds that operations at the proposed 7.5% uprated power level will have little impact on the operation of the SFP cooling system. This finding is based on the staff’s review and the experience gained from its review of power uprate applications for similar PWR plants.

5.2.2 Spent Fuel Pool Purification System

The purification system maintains the clarity and purity of the water in the fuel pool, refueling cavity, and refueling tank. It consists of the fuel pool purification pump, ion exchanger, filters, strainers, and an installed connection for a floating skimmer. The fuel pool pump circulates the fuel pool water through the filter and through the ion exchanger to remove particulate and ionic species. The purification system is normally operated on an intermittent basis when required by the fuel pool water conditions. Since the power uprate will not increase water temperature beyond that considered in the previous analyses, no damage to the ion exchange resin will occur. Also, the power uprate will not cause appreciable increase of fission products release from the fuel, and the removal of impurities from the pool water will not be affected.

The staff has reviewed the licensee’s evaluation and on the basis of its review finds that the purification function of the SFP will be adequate for maintaining purity of the SFP water after power uprate.

5.3 Service Water System

The SWS consists of two Seismic Category I flow paths and two non-Seismic Category I flow paths. The Seismic Category I flow paths supply two trains of cooling water to engineered safety features equipment, as well as provide the alternate water supply for the emergency feedwater pumps and an emergency makeup supply to the SFP. Each train is capable of providing 100% of the required flow and only one loop is required for plant shutdown after any postulated accident condition. The non-Seismic Category I flow paths are the auxiliary cooling water (ACW) system, which cools various non-safety related loads, and the path providing cooling water to the component cooling water system heat exchangers and the main chillers.

The licensee evaluated the effects of the proposed 7.5% power uprate on the SWS. The stator water coolers, which impose a load on the ACW system, will be replaced with larger-capacity coolers to allow an increase in the main generator electrical rating to support the power uprate. No changes are required for the Seismic Category I portion of the system. The licensee concluded that the SWS provides adequate cooling at the power uprate conditions.

The staff finds that operations at the proposed 7.5% uprated power level will have little impact on the operation of the SWS. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.4 Ultimate Heat Sink (UHS)

The ANO-2 UHS consists of two independent water sources, the emergency cooling pond (ECP) and the Dardanelle Reservoir. The Dardanelle Reservoir is the primary UHS during normal operations with the ECP as the Seismic Category I backup source for plant shutdown under normal and accident conditions. The design basis for the ECP is a LOCA in Unit 2 with a concurrent normal shutdown of Unit 1. The UHS was evaluated as part of the containment building design pressure increase review. The licensee performed its analysis for the containment building design pressure increase at the proposed uprated power. In an August 16, 2000, supplement to the November 3, 1999, Containment Uprate License Amendment Request, the licensee confirmed that the heat load rejected to the ECP will not increase above what is currently assumed in the AOR, even with the 7.5% power uprate. The staff determined that UHS would perform its intended function following a design-basis accident (DBA) in the safety evaluation associated with License Amendment 225, dated November 13, 2000.

5.5 Containment Cooling

The CCS maintains the ambient air temperature in containment during normal operation at or below the design-basis value of 110 ^{o}F. The safety-related portion of the system also reduces the containment post-accident pressure and temperature during a DBA. The CCS uses two groups of two cooling units, and each unit is equipped with two sets of coils and a vaneaxial fan located in containment outside the secondary shielding. A non-safety-related set of coils is cooled by chilled water for normal use and a safety-related set is cooled by service water for emergency cooling following a DBA.

As discussed in a June 29, 2000, submittal for License Amendment 226, the licensee performed new LOCA and MSLB analyses for the SG replacement that determined the peak containment pressure would increase from 54 psig to 58 psig. This analysis for the containment building design pressure increase was performed at the proposed uprated power. Under the new DBA conditions, the containment cooler fan blade pitch needed to be changed to keep the fan motor horsepower to its existing design capacity. The resultant modification reduced the cooling capacity of a given cooling unit by about 35%. Originally, only one cooling unit per group was required to be operable, but after the change in fan blade pitch, two cooling units per group are currently required by TSs to be operable to ensure the heat removal assumptions in the analysis are bounded. The staff determined that the CCS would perform its intended function following a DBA in the safety evaluation associated with License Amendment 226, dated November 13, 2000.

In its December 19, 2000, power uprate application, the licensee noted that the chilled water cooling coils were replaced with larger coil banks in refueling outage 2R14, which improved their heat transfer capability. The larger coil banks were installed to compensate for the reduction in fan blade pitch discussed above. The licensee determined that the CCS is now adequate for normal operations at the proposed uprated power level. The staff did not review the impact on plant operations at the proposed uprated power level on the design and performance of the containment chilled water cooling coils as they perform no safety-related function. Additionally, their failure to perform will not affect the performance of any safety-related system or component.

5.6 Turbine Generator

The high-pressure turbine steam path components and four stages of the low-pressure turbine were replaced with newer designs when the SGs were replaced in the last refueling outage (2R14). All replacement parts were provided by General Electric, the original equipment manufacturer. The turbine components were replaced to support operations with the RSGs and to optimize performance at the proposed uprated power. The design flow rate at the valve-wide-open condition ensures the turbine maximum guaranteed rating is met. The turbine trip functions and devices, including the overspeed trips, were determined to adequately protect the high- and low-pressure turbines at the proposed uprated power conditions. The licensee also evaluated the turbine generator auxiliary systems and determined they were adequate for the power uprate.

The licensee evaluated the impact of turbine modifications made for the power uprate on the generation of turbine missiles. The current analysis in the SAR states the generation of missiles due to turbine failure at or near normal speed and due to an overspeed incident is not credible. This is due, in part, to the reliability and redundancy of overspeed protection systems and the turbine construction. The licensee concluded that the modifications to the turbine did not increase the probability of missile hazards. Therefore, the licensee determined that the current analysis concerning the generation of turbine missiles is still valid.

The staff did not review the impact of plant operations at the proposed uprated power level on the design and performance of the turbine generator as it performs no safety-related function. Additionally, the failure of this system will not affect the performance of any safety-related system or component.

5.7 Main Steam Supply System

The licensee evaluated the impact of the power uprate on the main steam supply system and, where necessary, modified or replaced components. The major components of the main steam supply system include the MSIVs, the main steam safety valves (MSSVs), and steam supply line to the emergency feedwater turbine-driven pump. The power uprate will result in a main steam flow increase of about 1.1 million lbm per hour or about 9%. The most significant modification to the main steam supply system listed by the licensee for power uprate considerations, except for replacing the SGs, were modifications to the moisture separators-reheaters (MSRs). In refueling outages 2R12 and 2R13, modifications were made to the MSR chevrons and heater bundles. Heating and moisture removal at uprated power conditions were included in the design of the modifications to the MSRs. The licensee concluded that the components in the main steam supply system will perform their intended functions as described in the SAR at the proposed uprated power for normal, transient, and accident conditions.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the operation of the main steam system. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.8 Steam Dump and Bypass System (SDBS)

The licensee evaluated the effects of the proposed 7.5% power uprate on the SDBS and determined that no modifications to the SDBS are required for power uprate. The licensee also determined that the SDBS control system needed adjusting to accommodate the higher steam flow and change in steam header pressure versus power level. The licensee concluded that once the changes to the SDBS control system are made, the SDBS will adequately respond to transient conditions at the proposed power level.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the operation of the SDBS. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.9 Condensate and Feedwater System (CFWS)

The CFWS draws water from the condenser hotwells and high-pressure feedwater heater drains, and pumps it into the SGs. The CFWS consists of two interconnected trains of condensate pumps, heater drain pumps, main feedwater pumps, and feedwater heaters. The licensee evaluated the hydraulic and thermodynamic effects of operations at the proposed uprated power on the CFWS. The licensee concluded that, with the exception of the heater drain pumps, the components of the CFWS are adequate for operations at the proposed uprated power. The licensee's analysis determined that the existing heater drain pumps and motors did not have adequate capacity for operations at the higher power. As a result, three of the ten stages of the heater drain pumps will be replaced and higher horsepower pump motors will be installed, with a resulting increased capacity of approximately 9.5%.

The staff did not review the impact of plant operations at the proposed uprated power level on the design and performance of the CFWS, as it performs no safety-related function. Additionally, the failure of this system will not affect the performance of any safety-related system or component.

5.10 Emergency Feedwater System

The EFWS consists of a safety-related turbine-driven pump, a safety-related motor-driven pump, and two independent feedwater trains. The licensee evaluated the effect of the proposed 7.5% power uprate on the EFWS. The licensee determined that no modifications to the EFWS are required, as the current EFWS pumps can provide the minimum flow rate assumed by the safety analyses. The minimum flow rate used by the safety analyses is dependent on SG pressure, but is conservatively around 300 gallons per minute (gpm). The emergency feedwater pumps are designed to deliver 575 gpm at 1600 psia. The preferred source of water to the EFWS is the condensate storage tanks; the assured source is the SWS. In an October 12, 2001, response to an RAI, the licensee confirmed that the minimum TS volume of 160,000 gallons for the operable condensate storage tank is adequate to meet design condensate requirements. In a May 30, 2001, response to an RAI, the licensee verified that the SWS can provide adequate water for the EFWS.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the operation of the EFWS. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.11 Other Balance-of-Plant Evaluations

The impact of the proposed power uprate on the following systems and topical areas was evaluated by the licensee and found to be acceptable:

- condenser, cooling tower, and circulating water
- heater drain tank sizing
- auxiliary feedwater performance
- SG blowdown capacity
- carbon dioxide, nitrogen, and hydrogen systems
- instrument air system
- radwaste and primary sampling systems
- turbine building sump, oily waste, auxiliary building drains and sump
- auxiliary building heating, ventilation, and air conditioning (HVAC)
- turbine building HVAC

The staff finds that operations at the proposed power uprate will have little impact on the operation of the BOP systems and topical areas listed above. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.12 Containment Response Analysis

The licensee evaluated the LOCA and MSLB at the proposed uprated power to ensure the continued acceptability of the response of the containment structure. The analysis was performed in conjunction with the containment building design pressure increase review done to support the installation of RSGs. The analysis determined that the calculated peak accident pressure would increase from 54 psig to 58 psig. This review was documented in the licensee's November 3, 1999, submittal, which requested an increase in the containment building design pressure from 54 psig to 59 psig. That submittal was eventually supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 2000. The licensee performed its analysis for the containment building design pressure increase at the proposed uprated power. The staff found in a safety evaluation associated with License Amendment 225, dated November 13, 2000, that the licensee's calculations for peak accident pressure from a LOCA and a MSLB were acceptable. This acceptance was based on the fact that the licensee used NRC-approved calculational methods to perform the LOCA and MSLB calculations. In addition, a staff contractor performed audit calculations of the LOCA and MSLB accident with results similar to those performed by the licensee. License Amendment 225 also approved the increase in the containment building design pressure and set a TS limit for the maximum containment internal pressure during normal operation at 15.5 psia. The staff found that the licensee's analysis methods, assumptions, and conclusions for the containment building design pressure increase were based on sound structural engineering principles and acceptable ASME

Code requirements, and that the results of the accident calculations performed with the 15.5 psia TS limit satisfied all regulatory criteria.

5.13 Post-Loss-of-Coolant Accident Hydrogen Generation

The licensee evaluated the containment post-LOCA generation of H_2 at 3087 MWt (the proposed uprated power plus 2%). Its evaluation took into the account the increased reactor thermal power and the changes in containment temperature and pressure profiles due to the earlier containment uprate. The increase in power resulted in a 6.5% increase in the radiolytic decomposition generation of H_2 throughout the 30 days of the analysis period. The licensee did slightly modify the calculation for the total H_2 generated by the Zr-water reaction to use the weight of zircaloy specified in the current fuel cycle design document versus a slightly higher assumed value. The change in the estimated amount of H_2 generated was minimal. New metal corrosion rates were developed due to the change in the containment temperature and pressure profiles. The licensee also evaluated the change in the containment spray pH profile due to changing the buffering solution from sodium hydroxide to tri-sodium phosphate. The licensee determined that there would be an increase in H_2 production very early in the event due to corrosion of metals in containment, but a decrease in the total H_2 production over the 30 days of the analysis.

The licensee determined that without consideration of H_2 recombiner operations, the overall generation of H_2 over the 30 days of the analysis decreased by about 5%. The time required to reach a H_2 concentration of 2%, where the EOPs require startup of the H_2 recombiners, was unchanged. The analysis assumed the H_2 recombiners are started at a H_2 concentration of 3.5% and the time to reach that concentration decreased from 3.9 days to 3.5 days. The peak H_2 concentration, assuming one H_2 recombiner operating, increased from 3.5% to 3.8%, and the H_2 concentration after 30 days increased from 2.3% to 2.4%.

The licensee concluded that the H_2 recombiner system is capable of maintaining containment H_2 below the 3.9% design limit at the uprated power. The licensee also determined that no changes to operator actions taken in response to a LOCA for combustible gas control in containment are necessary.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the generation of H_2 in containment after a LOCA. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.14 High Energy Line Break

The licensee had previously revised the HELB analysis for breaks outside the containment for the RSG configuration change. The analyses were performed at the time of the RSG configuration change during refueling outage 2R14 and were performed at the proposed uprated power levels. Any changes in the HELB analysis were incorporated into the SAR under 10 CFR 50.59 in SAR Amendment 16. The licensee did not perform any additional analyses for the power uprate amendment. The systems identified in the SAR as those in which design-basis piping breaks occur are:

- main steam
- main feedwater

- SG blowdown
- chemical and volume control charging and letdown
- emergency feedwater

The licensee determined that the proposed power uprate will have little impact on the consequences of a HELB outside of containment.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the consequences of HELB outside of containment. This finding is based on the staff's review of SAR Section 3.6 and the experience gained from its review of power uprate applications for similar PWR plants.

5.15 Control Room Uninhabitability

Control room uninhabitability refers to the provisions made to allow operators to maintain the plant in a safe shutdown condition from outside the control room following a control room evacuation. The licensee determined that there was no impact due to the proposed power uprate on the remote shutdown facilities after a control room evacuation for a non-Appendix R event.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on control room uninhabitability. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.16 Fire Protection Program

The licensee has conducted substantial reviews of the impact of the proposed power uprate on the Fire Protection Program by reviewing the impact of plant modifications associated with the proposed power uprate and the overall program impact. No necessary changes to the fire protection systems, with one exception, have been identified so far. The exception was one case where the penetration seal material needed to be replaced due to increased main steam pipe temperatures. The licensee is finalizing the reviews of the plant modifications to be done during refueling outage 2R15. No further fire protection modifications are expected to result from the remaining reviews. The licensee concluded that the plant modifications will not adversely affect the ability to achieve safe shutdown in a fire scenario. The alternate shutdown procedure was determined to be adequate with no changes.

The staff finds that operations at the proposed 7.5% uprate power level will have little impact on the fire protection program. This finding is based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants.

5.17 Flow-Accelerated Corrosion

The power uprate will result in an increase of the flow rates in certain systems in the plant. This increase may affect FAC in these systems. The licensee estimated this effect by performing a parametric study using the predictive code CHECWORKS. The study included the following systems that are in the FAC program: main steam, main feedwater, reheat steam, high-pressure extraction, low-pressure heater vents and drains, high-pressure heaters vents and drains, condensate, and SG blowdown. The licensee incorporated into the CHECWORKS code the most recent inspection data, in order to provide a baseline for comparing current wear predictions with the wear rates resulting from the changes in operating conditions caused by power uprate. The parametric study revealed that the worst operating conditions for the uprated power conditions had a minimal impact on the FAC wear rates. The licensee concluded, therefore, that there was no need for physical modification of the plant and only minor adjustment to the FAC model in the CHECWORKS code will be required.

The staff has accepted CHECWORKS as a conservative predictive model for predicting loss of material due to FAC. The staff reviewed the methodology used by the licensee and the licensee's parametric FAC analysis for the uprated power conditions. The staff has determined that the licensee's parametric CHECWORKS analysis for the carbon steel components in the ANO-2 plant under the uprated power conditions applied conservative assumptions for input parameters. The staff, therefore, concludes that the licensee's FAC program for the unit will continue to provide assurance that the licensee will detect significant FAC-induced degradation prior to failure of a carbon steel component that is important to safety, even under the uprated conditions for the facility. The staff concludes that the licensee's FAC program for ANO-2 will continue to provide an acceptable means of monitoring for FAC-induced degradation, even under the uprated conditions for the facility.

5.18 Electrical Systems

The proposed power uprate will revise the operation of the ANO-2 reactor power level from 2815 MWt to 3026 MWt (approximately 7.5%). The proposed change would increase the unit's design gross electrical output from approximately 958 MWe to approximately 1065 MWe.

ANO-2 generates electric power at 22 kV which is fed through an isolated phase bus to the main transformer bank. The main transformer bank, consisting of three single-phase transformers, steps the output voltage to 500 kV and connects to the station switchyard. The 500 kV switchyard is a 2-bus design consisting of a breaker-and-half scheme. The 500 kV switchyard includes three outgoing lines.

A bus tie autotransformer bank consisting of three single-phase autotransformers interconnects the 500 kV and 161 kV systems in the switchyard. There are two outgoing lines from the 161 kV switchyard. The 22 kV tertiary winding of the autotransformer bank supplies startup transformer (SAT)-3, which is identical to the unit auxiliary transformer (UAT). SAT-2, which serves both units, is supplied from the 161 kV ring bus. Auxiliary power for normal plant operation is supplied by the main generator through the UAT. Auxiliary power for main generator startup and shutdown can be supplied by either of the two SATs. In the event of non-availability of these two power sources, power to the engineered safety features buses can be supplied by the two fully redundant emergency DG sets.

The offsite transmission system, and the onsite distribution system are designed to provide electric power for all electrical equipment for startup, normal operation, safe shutdown, and handling of all emergency situations. The electrical distribution system has been previously evaluated to conform to 10 CFR Part 50, Appendix A, GDC-17.

The following is an evaluation of grid stability, the main generator, the transformers, and the emergency DGs.

5.18.1 Grid Stability

The licensee has evaluated the transmission system to ensure that the grid remains stable for the upgraded generation level at ANO. Increasing the generation level results in a slight reduction in stability margins. However, this slight reduction in margins does not result in instability for the disturbances expected on the transmission system. During such disturbances, the offsite system will continue to supply the safety related buses with acceptable voltage levels.

The licensee performs transmission studies every two years to reconfirm the acceptability of the offsite power sources. These studies ensure that the grid stability is retained during and after a single contingency. The licensee’s December 19, 2000, application indicated that the next transmission studies were scheduled for the first quarter of 2001. A recent transient stability study confirmed the stability of the offsite power system during summer peak conditions following the simultaneous loss of both ANO units. A generated output for the station was assumed that bounds the ANO-2 power uprate. This study indicated that a three-phase fault near the ANO switchyard, on any of the three 500 kV lines leaving ANO (cleared under a breaker trip time of 5 cycles), results in the ANO units and the system remaining stable for both existing and power uprate generation levels. The study indicates that, for both current and uprate generation levels, when either the Mablevale or the Pleasant Hill 500 kV line is already out of service, the ANO units are unstable following a three-phase fault on the other of these two lines. To avoid this situation, procedures require that if one of these two 500 kV lines is out of service, the ANO generation for both units is reduced to 1300 MWe to prevent instability should a fault occur on the other of these two lines. Additionally, to avoid unit instability during periods of minimum load conditions on the transmission system, ANO procedures limit the ANO-2 absorption levels to 200 mega-volt-amp-reactance (MVAR). Grid stability studies have demonstrated that for power uprate, the transmission grid remains stable and the safety-related buses will be acceptably supplied by the offsite power sources following postulated transmission system disturbances. The staff concluded that ANO continues to meet GDC-17 for grid stability with this power uprate.

5.18.2 Main Generator

The main generator is currently rated at 1047.3 mega-volt-amp (MVA) at 0.90 power factor (lagging), which equates to 942 MWe and 457 MVAR. Although the generator stator was rewound for power uprate conditions, the generator rating was not increased due to limitations in the stator water system. From capability curves prior to rewind, the generator was limited to approximately 370 MVAR (lagging), that equated to 0.936 power factor. The Entergy Transmission and Distribution group has restricted the power factor to be 0.95 or higher at full generator capability. The licensee will install large stator water coolers, allowing the generator rating to be increased from 1047.3 MVA to 1133.3 MVA at 0.95 power factor that equals to 1077 MWe and 353 MVAR. According to revised GE capability curves for the rewound generator, the generator is limited to approximately 362 MVAR (lagging) at 1065 MWe. This equates to a 0.95 power factor. Therefore, from these considerations, operating it at the uprated power condition is acceptable.

5.18.3 Main Power Transformer

The main transformer bank consists of three single-phase units, each rated at 333.5 MVA, for a total capacity of 1000.5 MVA. At power uprate conditions, approximately 4% (41.5 MWe) of the power will flow into the unit auxiliary transformer to serve ANO-2 house loads, and 1023.5 MWe will flow through the main power transformer.

Section 2.2.3.2 of Enclosure 5 of the December 19, 2000, application stated that additional main power transformer cooling will be installed during refueling outage 2R15. Contrary to that statement, in the October 30, 2001, supplemental letter, the licensee states that this modification has been deferred and will not be installed during refueling outage 2R15. The decision to defer this modification was based on the scheduling and commercial aspects associated with refueling outage 2R15. The licensee states that a significant amount of life remains in these transformers, since they have operated at less than rated capacity since installation. The calculated insulation remaining life per Institute of Electrical and Electronics Engineering (IEEE) Standard C57.91, "IEEE Guide for Loading Mineral-Oil-Immersed Transformers," is still greater than 90%.

After reviewing the October 30, 2001, supplemental letter, the staff submitted an RAI for supporting documentation that justifies that the main transformers are more than capable of operating at uprated conditions due to extra capacity designed into these transformers. By supplemental letter dated November 16, 2001, the licensee stated that, following the power uprate, the loading on the ANO-2 main power transformer will be above the 100% nameplate rating, which is 1000.5 MVA (forced oil and air (FOA) @ 65 ^{o}C average winding temperature rise above maximum ambient temperature of 40 ^{o}C and maximum 24-hour average ambient temperature of 30 ^{o}C). To address the acceptability of the main transformers at elevated loadings, the licensee funded a study for the transformer design. The study indicated that these transformers are capable to withstand loads of 107% of their rating without cooling and 113% of their rating with cooling. During the summertime conditions, the load is expected to be approximately 101% to 103% of the main transformers' existing 1000.5 MVA rating. During the wintertime conditions, the main transformers can be loaded to 105% to 109% of the 1000.5 MVA transformer rating. The potential 109% loading during the wintertime months is more limiting and is acceptable since the ambient temperatures at these times are well under the rated 24-hour average temperature of 30 ^{o}C. Per IEEE Standard C57.91-1995, Table 4, the loading on FOA transformers can be increased by 0.75% for every degree Centigrade the average ambient temperature is lowered from the rated 30 ^{o}C. During the wintertime, when the main transformers could be operated at 105% to 109% of their rating, the average ambient temperature will be less than 20 ^{o}C. Consequently, the proposed 105% to 109% loading, which will only occur during the wintertime, is acceptable.

On the basis of this information, the staff finds that the existing transformer coolers continue to perform acceptably at this time and these coolers will keep the transformers below their rated temperatures at uprate power conditions.

5.18.4 <u>Unit Auxiliary Transformer</u>

The unit auxiliary transformer is a three-winding transformer having an overall capacity of 58.5 MVA, with a capacity of 32.8 MVA on the 6.9 kV winding and a capacity of 25.7 MVA on the 4.16 kV winding. Power uprate will not increase the load on 6.9 kV winding. The expected increase due to power uprate is about 0.70 MVA at 0.89 power factor, and brings the 4.16 kV winding loading to about 80% of capacity. Therefore, from these considerations, operating at the uprated power condition is acceptable.

5.18.5 Startup Transformer-3

SAT-3 is identical to the ANO-2 unit auxiliary transformer in size and rating. This transformer is used for startup and shutdown. The minor load added by the power uprate is not significant to cause any transformer winding overload. No changes are required for power uprate. Therefore, operating at the uprated power condition is acceptable.

5.18.6 Startup Transformer-2

SAT-2 is a three-winding transformer having an overall capacity of 45 MVA, with a capacity of 25 MVA on the 6.9 kV winding and a capacity of 21 MVA on the 4.16 kV winding. It is the second (delayed access) of the two required offsite power sources needed for ANO-2 accident mitigation. It is shared by ANO-1 and ANO-2. The equipment has been evaluated for power uprate and the licensee has procedures in place to ensure the rated capacity of the transformer will not be exceeded. Therefore, operating at the uprated power condition is acceptable.

5.18.7 Isolated Phase Duct

The isolated phase duct is designed to accept the maximum generator output and, therefore, will continue to support plant operations at uprate conditions.

5.18.8 Emergency Diesel Generators

There is no change to the safety-related loads at uprate conditions and, therefore, the emergency DGs will not be impacted by power uprate and will remain capable of performing their safety-related functions during a LOOP/LOCA and power uprate.

5.18.9 Electrical Systems Summary

The staff has evaluated the effect of power uprate on the necessary electrical systems. Results of these evaluations show that the increase in core thermal power would have negligible impact on the grid stability. This is consistent with GDC-17 and the proposed change is, therefore, acceptable.

6.0 HUMAN FACTORS

The staff reviewed the operator performance aspects of the application and discussed five topic areas during an April 25, 2001, telephone conference with the licensee. In a June 20, 2001, supplemental letter, the licensee responded to the five questions related to operator performance. The questions and the evaluation of the licensee's response follows:

6.1 Changes in Emergency and Abnormal Operating Procedures (AOPs)

> Describe how the proposed power uprate will change the plant emergency and abnormal procedures.

The licensee stated in its letters dated December 19, 2000, and June 20, 2001, that the "power uprate has no effect on the type and scope of the ANO-2 emergency and abnormal operating procedures (EOPs and AOPs). No new operator actions are needed for power uprate and the type and nature of operator actions needed for accident mitigation will not change." New procedures will not be required for the uprate. Certain setpoints will be revised to be consistent with the power uprate analyses.

The staff finds that the licensee's response is satisfactory because the procedures will be revised to incorporate the new setpoint values prior to implementation for the power uprate. The licensee will treat plant procedure changes in a manner consistent with any other procedure changes.

6.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

> Describe any new risk-important operator actions required as a result of the proposed power uprate. Describe changes to any current risk-important operator actions that will occur as a result of the power uprate. Explain any changes in plant risk that result from changes in risk-important operator actions. (e.g., Identify operator actions that will require additional response time or will have reduced time available. Identify any operator actions that are being automated as a result of the power uprate. Provide justification for the acceptability of these changes.)

The licensee stated in its supplemental letter dated June 20, 2001, that, "there were no changes made to operator action assumptions in [licensing-basis] accident or transient analyses that resulted in reduced operator response times. There are no new operator actions due to power uprate." Any required setpoint changes due to the power uprate values for design parameters such as thermal power, decay heat, and temperatures will be incorporated into the procedures prior to power uprate. These procedures will be demonstrated during the verification and validation phase of the changed procedures.

The staff finds that the licensee's response is satisfactory because the licensee has adequately addressed the question of operator actions sensitive to the power uprate. The licensee will reevaluate and update procedures as required for the power reductions required to respond to failure of components like the main feedwater pump or the circulating water pump.

6.3 Changes to Control Room Controls, Displays, and Alarms

> Describe any changes the proposed power uprate will have on the operator interfaces for control room controls, displays and alarms. For example, what zone markings (e.g., normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed power uprate and how operators were tested to determine they could use the instruments reliably.

The licensee stated in its supplemental letter dated June 20, 2001, that, “changes to operator interfaces for control room controls, displays and alarms necessitated by power uprate will be minimal." No instruments are being upgraded from analog to digital instruments as a result of power uprate. No physical modifications are being made to control stations. The plant computer software will be revised so the range is consistent with the higher power rating. New turbine-generator load meters were already replaced with new scales that can read up to 1200 MWe to allow adequate margins. Other examples of changes include: the control board indicators for the turbine first-stage pressure and the intermediate-range pressure transmitter were modified to accommodate power uprate range requirements, the calibrated range of the steam and feedwater flow transmitters was increased, and the control board recorder chart paper was rescaled. Power uprate will necessitate various setpoint changes. However, such changes do not directly affect the operator's interface with the existing control stations.

The staff finds that the licensee’s response is satisfactory because the licensee has adequately identified and described the changes that will occur to displays and controls as a result of the power uprate.

6.4 Changes to the Safety Parameter Display System

> Describe any changes the proposed power uprate will have on the Safety Parameter Display System. How will the operators know of the changes?

The licensee indicated that changes to the SPDS will be minimal and will be implemented in accordance with standard engineering and software control procedures. The format for SPDS displays will not change as a result of power uprate.

The staff finds that the licensee’s response is satisfactory because the licensee will identify changes that will occur to the SPDS as a result of the power uprate.

6.5 Changes to the Operator Training Program and the Control Room Simulator

> Describe any changes the proposed power uprate will have on the operator training program and the plant reference control room simulator, and provide the implementation schedule for making the changes.

The licensee's response in its respective application and supplemental letters dated December 19, 2000, and June 20, 2001, indicates that simulator modifications will be incorporated according to the standard process for maintaining the simulator. The licensee listed the uprate changes that affect specific simulator process models. After the process model modifications are complete, simulator operability as described in American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.5 will be conducted for changes required by the power uprate.

The ANO-2 training staff will provide classroom training and simulator training on the power uprate changes prior to refueling outage 2R15.

The staff finds the licensee's response satisfactory because the licensee has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes in accordance with the requirements of ANSI/ANS-3.5.

6.6 Human Factors Evaluation Summary

The staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

7.0 RADIOLOGICAL ANALYSIS

7.1 Atmospheric Relative Concentration Estimates

The licensee used five years of onsite meteorological data collected during calendar years 1995 through 1999 to estimate the atmospheric relative concentration (X/Q) values used in the control room dose assessments described in Section 7.3 below. These data were measured at 10 meters and 57 meters above grade at the ANO-2 site. The licensee confirmed that the meteorological measurement program meets the recommendations of RG 1.23, "Onsite Meteorological Programs." The tower components were tested per surveillance test procedures once every six months and system calibrations were performed. Problems found were promptly resolved in compliance with the plant corrective action program. The licensee performed a limited comparison of historical data with the data collected from 1995 through 1999 and noted that the results were comparable. The licensee also noted that the meteorological tower area was free of obstructions during the 1995 through 1999 measurement period.

The staff performed a review of the data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," on meteorological data quality assurance. The overall joint data recovery of reported wind speed, wind direction, and atmospheric stability exceeded 95% at both levels every year and averaged 98% for the five-year period. Thus, reported data recovery surpassed the recommended minimum of 90% cited in RG 1.23. The longest continuous period of data outage was approximately four days. Further examination of the data indicated that temperature differences used to estimate stability appear somewhat different than expected, when compared to an open site with homogenous flat terrain. These indications include the occurrence of a relatively large number of extremely and moderately unstable conditions (Pasquill-Gifford stability classes A and B) during the day in winter. These indications also include occurrences, although infrequent, of stability classes A and B reported during the night, and F and G stability classes (moderately and extremely stable lapse rates) during mid-day. Further, A and B stability classes were each infrequently reported to occur for relatively long periods of time; as long as 16 consecutive hours and 21 consecutive hours, respectively. Thus, the ANO-2 measurements may be affected by local changes in surface conditions (e.g., topography, surface roughness, thermal characteristics). A comparison of the five-year period of data, with one year of historical data from February 1972 to February 1973, indicated a higher occurrence of stability class A and unstable conditions, as well as a decrease in wind speeds in the 1995 through 1999 data period. The licensee noted that the meteorological measurement program was upgraded in 1986. For the 1995 through 1999 data, comparison of wind direction showed good year-to-year correlation at both measurement heights, and between the lower level and upper level measurements.

In their dose assessment, the licensee used design basis input X/Q values for the exclusion area boundary (EAB) and low-population zone (LPZ) as reported in the SAR at the time ANO-2 received its operating license in 1978. The LPZ values are slightly higher than those currently in Section 2.3 of the updated SAR because they are calculated for a shorter LPZ distance than the original proposed distance of four miles. These X/Q values are based on a methodology that is somewhat different than current NRC guidance and staff practices. For example, for the 0-8 hour time period, the licensee used the X/Q equation in RG 1.4, "Assumptions Used for

Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," and selected the 5 percentile X/Q values as a function of direction. However, in the staff's opinion, the RG 1.4 methodology assumes that X/Q values are independent of direction and, therefore, selection of 5 percentile X/Q values by direction would be inappropriate.

RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," issued in 1979, recommends use of the higher of the 0.5 percentile X/Q value by direction or the 5 percentile X/Q value for the overall site. The staff performed independent calculations with resultant X/Q estimates somewhat higher than those used by the licensee in their dose assessment. However, because the resultant dose at the LPZ is still within the 10 CFR Part 100 limit, the staff has concluded that there is reasonable assurance of adequate protection of the public for this amendment. Thus, this is not a safety issue for this amendment and use of these assumptions will be reviewed on a case-by-case basis for future applications.

The licensee used the ARCON96 methodology (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake") to calculate X/Q values for control room dose assessment. Calculations for the LOCA were made assuming a release from the containment surface for the shortest distance between the containment surface and the control room intake, and from the containment flute for the ECCS portion of the release. The limiting X/Q value was then used in all of the dose assessments for the postulated LOCA. Calculations were also made for the fuel handling accident (FHA). The X/Q values were calculated as ground level releases and assumed no effluent flow. The postulated release directly from the containment building was assumed to be diffuse. In addition, for the SGTR and CEA ejection dose assessments (secondary side), the licensee used control room X/Q values previously approved as part of Amendment 222, dated September 29, 2000.

The following X/Q values are acceptable for this amendment, but staff finds the methodology used by the licensee to calculate the EAB and LPZ X/Q values deficient, since it selects 5 percentile values as a function of direction. The staff has determined that it is not necessary to revise the calculations in support of this amendment since 10 CFR Part 100 dose limits can still be met using a direction-dependent model, with selection of the higher of the 0.5 percentile X/Q value by direction or the 5 percentile X/Q value for the overall site. However, future license amendments involving dose assessments will be reviewed on a case-by-case basis.

Table 7.1 - Arkansas Nuclear One, Unit 2, Atmospheric Relative Concentration Values		
Offsite X/Q Values for LOCA, CEA, FHA and SGTR		
0 - 2 hour EAB	6.5 E-4 seconds per cubic meter (s/m^3)	
0 - 8 hour LPZ	3.1 E-5 s/m^3	
8 - 24 hour LPZ	3.6 E-6 s/m^3	
1 - 4 day LPZ	2.3 E-6 s/m^3	
4 - 30 day LPZ	1.4 E-6 s/m^3	
Control Room X/Q Values (LOCA, including ECCS, and CEA from Containment Flute)		
0 - 2 hour	9.77 E-4 s/m^3	
2 - 8 hour	5.76 E-4 s/m^3	
8 - 24 hour	2.56 E-4 s/m^3	
1 - 4 day	1.68 E-4 s/m^3	
4 - 30 day	1.25 E-4 s/m^3	
FHA from Fuel Handling Flute		
0 - 2 hour	1.2 E-3 s/m^3	
SGTR and CEA Ejection - Secondary Side		
	Atmospheric Dump Valves	MSSVs
0 - 2 hour	6.31 E-4 s/m^3	8.05 E-4 s/m^3
2 - 8 hour	3.65 E-4 s/m^3	4.64 E-4 s/m^3

7.2 Control Room Habitability

The licensee's July 3, 2001, supplemental letter provided calculations of the postulated consequences to the control room operators of the FHA, maximum hypothetical accident (MHA) (which is the LOCA), SGTR, and CEA ejection accidents. These calculations were based upon no unfiltered inleakage into the control room envelope and 10 cubic feet per minute (cfm) of inleakage due to ingress/egress into the envelope.

In November 2001, tracer gas tests of the control room envelope were performed by a licensee's contractor to determine the control room envelope's inleakage characteristics. Four tests were performed. All showed unfiltered inleakage being greater than the assumed value of zero. The maximum inleakage was 124 cfm.

The tests identified the principal sources of and conditions conducive to inleakage. The licensee's January 14, 2002, supplemental letter to the staff identified the VSF-9 fan and housing (45 cfm) and the interface between the north wall of the control room envelope and the controlled access-2 (CA-2) room (49 cfm) as the major sources. The interface was a source of inleakage whenever both 2VEF-56 fans were operating in conjunction with VSF-9.

To reduce the inleakage, the licensee proposed to repair the seals on the doors and the housing of VSF-9. This same letter indicated that the repair of those seals were expected to be concluded by February 28, 2002.

The licensee conducted pressure sweeps which showed that the CA-2 area would not be at a positive pressure relative to the control room envelope when the 2VEF-56 fans were operating if the double doors from the CA-2 area to the turbine building were opened. The licensee proposed to implement administrative controls to open both doors from CA-2 to the turbine building prior to operating both 2VEF-56 fans.

The licensee projected that these two corrective actions would result in control room envelope inleakage being reduced to 30 cfm with an additional 10 cfm due to ingress/egress. In their January 14, 2002, supplemental letter, the licensee provided regulatory commitments to seal VSF-9 and to establish administrative controls to open the double doors from CA-2 to the turbine building prior to the actuation 2VEF-56 fans. This letter indicated that these commitments would be tracked through the licensee's corrective action program and commitment management system.

The July 3, 2001, supplemental letter contained a parametric analysis which indicated that the control room operators' thyroid dose following a MHA would approach the 30 rem limit when the combined unfiltered inleakage rate and the inleakage from ingress/egress was 61 cfm. In the January 14, 2002, supplemental letter, the licensee proposed to change the design and licensing basis and increase the unfiltered inleakage including ingress/egress to 61 cfm. In a supplemental letter dated January 31, 2002, the licensee also proposed, as a regulatory commitment, maintenance of the control room envelope's inleakage to less than 61 cfm.

The staff has performed an independent assessment of the control room operators' doses for the FHA, MHA, SGTR, and CEA ejection accidents for an unfiltered inleakage of 61 cfm. These results are presented in Tables 7.3-5 and 7.3-6. The staff calculated doses which showed doses less than the acceptance criteria presented in SRP Section 6.4. Consequently, the control room operators' doses meet GDC-19 of Appendix A to 10 CFR Part 50.

Although the licensee switched the design-basis inleakage to 61 cfm, it did not submit revised calculations of the control room operators' doses for the FHA, the SGTR, or the CEA ejection accidents. The staff confirmed that the consequences of these accidents did not result in doses which would exceed GDC-19. All subsequent calculations of the radiological consequences of the FHA, SGTR, and CEA ejection accidents should include the doses to the control room, as the applicability of GDC-19 is not limited to only the MHA. Additional details on the control room operators' doses for the particular accident are contained in this safety evaluation for the particular accident.

7.3 Radiological Analysis

The licensee has performed re-analyses of the ANO-2 SAR Chapter 15 accidents. Re-analyses were required in order to incorporate the increase in core power and associated changes in various operating parameters. Certain accidents were re-analyzed as part of one of the RSG amendments (ANO-2 Amendment 222) and were not re-analyzed as part of the power uprate amendment. Accidents which fall into the latter category were:

1) Loss of External Load and/or Turbine Trip
2) Loss of Normal Feedwater Flow
3) Loss of All Normal and Preferred AC Power to Station Auxiliaries
4) Excess Heat Removal Due to Secondary System Malfunction
5) MSLB
6) Transients Resulting from the Instantaneous Closure of a Single MSIV

Accidents which result in a radiological release which were indicated in Table 7.3.0-1 of the December 19, 2000, application as being re-analyzed included the following:

1) Seized Rotor
2) MHA
3) FWLB
4) SGTR
5) CEA Ejection
6) FHA

Although the above accidents were noted as being re-analyzed as part of the power uprate amendment request, in fact, for the seized rotor and the FWLB, the radiological releases and associated doses aspects were not re-analyzed. The licensee's submittal in support of Amendment 222 had determined the radiological releases and associated doses at the uprated power level. The staff had reviewed those calculations during the review of Amendment 222. Since there have been no changes made since issuance of Amendment 222, no additional reviews were required by the staff.

Accidents for which calculations were reviewed by the staff for this amendment included:

1) MHA
2) SGTR
3) CEA Ejection
4) FHA

The licensee's calculations determined the whole body, skin, and thyroid doses to members of the public located at the EAB and the LPZ, and to the control room operators. The following sections provide the staff's assessment of the potential consequences of the FHA, the SGTR, the MHA, and the CEA ejection accidents.

7.3.1 Maximum Hypothetical Accident

For a MHA, the licensee assumed that the releases to the environment occurred via two pathways. The first was leakage from the reactor building. The second was via the containment flute and resulted from ECCS recirculation loop leakage.

As a result of the MHA, the fuel is assumed to melt with fission products being released to the reactor building. These releases are airborne in the reactor building but are reduced by the effect of engineered safety feature (ESF) sprays. The licensee assumed that operation of the reactor building cooling systems resulted in the distribution of airborne activity within the reactor building between the sprayed and unsprayed regions.

The licensee assumed that ECCS recirculation loop leakage was released with no credit for holdup or filtration. Because the airborne release resulting from ECCS leakage is not treated by an ESF ventilation system, the licensee assumed that a passive failure occurred at 24 hours following the MHA. This passive failure was assumed to result in a release of 5 gpm of recirculation fluid for 30 minutes. The licensee assumed that the partition factor for iodine for this release was 1. Following these 30 minutes, ECCS leakage is assumed to return to the level it was prior to the passive failure.

The staff assessed the potential consequences of a MHA. The assumptions which were incorporated into the staff's assessment are included in Table 7.3-1. The thyroid and whole body doses, which were calculated by the staff, are presented in Tables 7.3-5 and 7.3-6, respectively. The doses were found to be within the acceptance criteria associated with 10 CFR Part 100 and GDC-19 of Appendix A to 10 CFR Part 50, and therefore, acceptable.

It should be noted that the guidance in SRP Section 15.6.5 has the passive failure leakage rate being 50 gpm for 30 minutes rather than the 5 gpm assumed by the licensee. The partition factor for iodine in the SRP for this release is typically 0.1. Therefore, the licensee's assumption of 5 gpm with a partition factor of 1 is equivalent to the staff's assumption of a leakage rate of 50 gpm with a partition factor of 0.1. Consequently, although the licensee utilized a lower passive failure rate, the lower passive failure rate is acceptable because the licensee utilized a partition factor of 1.

7.3.2 Fuel Handling Accident

The licensee performed one bounding analysis to cover a refueling accident occurring in either the reactor building or the spent fuel building. In the analysis, it was assumed that the dropping of a spent fuel assembly resulted in damage to 60 fuel rods. The gap activity in these rods was assumed to be released instantaneously to the environment without treatment by filtration by an ESF filter ventilation system, or without decay due to holdup in the building. The licensee's analysis assumed that the equipment hatch and the personnel air locks in the reactor building were open during the event.

Table 7.3-2 contains details of the assumptions utilized by the staff in their assessment of the potential consequences of a FHA. The doses calculated by the staff are presented in Tables 7.3-5 and 7.3-6. The doses were found to be less than the acceptance criteria presented in SRP Section 15.7.5 and GDC-19 of Appendix A to 10 CFR Part 50 and therefore, acceptable.

Although the dose assessment assumed fuel handling operations with the containment equipment hatch and personnel air locks open and the doses were found acceptable, acceptable doses neither relieve nor exempt the licensee from meeting the fuel and radioactive release monitoring requirements of GDC-63 and GDC-64 of Appendix A to 10 CFR Part 50.

7.3.3 Rod Ejection

The licensee performed an analysis of a postulated CEA ejection accident. The licensee assumed that the accident occurred with the reactor operating at its TS limit for primary to secondary leakage (150 gallons per day (gpd) per SG). The CEA ejection accident was assumed to induce fuel failures. The licensee's analysis assumed that the CEA ejection accident resulted in fuel damage to 15% of the fuel rods in the core. This results in a release of the gap activity from the affected rods. No fuel melting was assumed to occur as a result of the CEA ejection accident.

Two potential release pathways were evaluated. The first assumed that the activity from the damaged fuel was released to the reactor building with subsequent release to the environment via leakage. The second pathway assumed that the activity from the damaged fuel was released to reactor coolant. From reactor coolant, activity reached the secondary side as a result of primary to secondary leakage and release to the environment occurred via the steam being discharged from the atmospheric dump valves.

Table 7.3-3 presents the assumptions utilized by the staff in their assessment. The potential consequences are presented in Tables 7.3-5 and 7.3-6. The doses were found to be less than the acceptance criteria noted in SRP Section 15.4.8 and GDC-19 of Appendix A to 10 CFR Part 50 and are, therefore, acceptable.

The staff noted that the calculations performed by the licensee assumed that the releases to the environment ceased at 8 hours following the CEA ejection accident. The staff believes that the licensee assumed the 8 hours based upon the time necessary to institute shutdown cooling. However, the staff did see within the licensee's calculations a basis for assuming that the reactor building leakage would only occur for 8 hours. While subsequent time periods result in small components of releases, the staff calculations showed the control room operators' dose for the CEA ejection accident to be greater than the dose due to the MHA when carried out to 30 days. Future CEA ejection accident containment assessments should have a basis for the period of release from the reactor building and the duration of control room operators' exposure.

7.3.4 Steam Generator Tube Rupture

The licensee evaluated the consequences of a postulated SGTR accident. The licensee's analysis assumed the SGTR occurred with primary to secondary leakage at 720 gpd (0.5 gpm) per SG rather than at the TS value of 150 gpd per SG. The licensee analyzed the SGTR at this primary to secondary leak rate because it bounds design-basis events, which create a large differential pressure between the primary and secondary systems due to the opening of the secondary side to atmosphere. The licensee considered the MSLB and FWLB as accidents which create such a large differential pressure.

The licensee analyzed two cases for the SGTR. The first case assumed a pre-existing spike prior to the tube rupture. For the pre-existing spike case, the reactor coolant iodine activity was assumed to be at 60 microcuries per gram (µCi/gm) of dose equivalent Iodine-131 (I-131). The secondary coolant iodine specific activity was assumed to be at the secondary coolant specific activity equilibrium value of 0.10 µCi/gm. The second case, referred to as the accident-initiated spike case, assumed the tube rupture initiated an iodine spike concurrent with the accident. Immediately prior to the accident, the reactor coolant was assumed to be at the TS value of 1.0 µCi/gm of dose equivalent I-131. Secondary system activity was again assumed to be at 0.10 µCi/gm dose equivalent I-131. The tube rupture was assumed to initiate an iodine spike which would result in a release of iodine from the fuel gap to the reactor coolant at a rate which is 500 times the normal iodine release rate necessary to maintain the reactor coolant activity level at 1.0 µCi/gm of dose equivalent I-131. The licensee indicated that the tube rupture did not result in any melted fuel being released to the reactor coolant.

For both cases, it was assumed that offsite power was lost and the main condenser was unavailable for the steam dump. The licensee's assessment assumed that the faulted SG would be isolated within 1.0 hour and that it took 8 hours following the tube rupture before residual heat removal could be initiated.

The licensee's isotopic distribution of iodine in the reactor coolant was based upon a core distribution. This was in lieu of the isotopic distribution provided in SAR Table 11.1-3.

The staff assessed the consequences of the SGTR. The initial assessment performed by the staff showed a significant variation between the licensee's results and the staff's. A part of the variation could be attributed to the staff assuming that the charging pump flow was equivalent to the letdown flow. If the charging pump flow is limited to 46 gpm, then accounting for the maximum unidentified and identified primary-to-secondary leak rates limits the letdown flow to 35 gpm. In a January 31, 2002, supplemental letter to the staff, the licensee indicated that charging pump flow was limited to 46 gpm.

A second reason for the discrepancies between the staff's and the licensee's results can be attributed to the licensee's selection of the isotopic distribution of the iodine isotopes. The licensee selected the isotopic distribution based upon the core distribution of iodine isotopes. The staff concluded that basing the distribution on core inventory was inappropriate. In discussions with the staff, the licensee agreed that their selection was probably inappropriate. The licensee has subsequently performed a re-calculation of the iodine activity levels in the primary coolant. The results of these re-calculations were provided to the staff in a January 31, 2002, supplemental letter. The new results more closely reflected the calculations performed by the staff. The licensee has not resubmitted the resultant dose assessments for these new isotopic activity levels. However, because the staff and the licensee are now in good agreement relative to the isotopic distributions and activity levels, and since the staff calculated EAB and LPZ doses which meet the acceptance criteria of SRP 15.6.3 and control room operators' doses which meet SRP 6.4, the staff has concluded that this provides sufficient justification to conclude that the consequences of a SGTR are acceptable. The licensee committed in a January 31, 2002, supplemental letter to the staff that subsequent analyses of the consequences of a SGTR will be based upon the equilibrium activity levels of the iodine isotopes in primary coolant and not the core isotopic distribution.

The calculations performed by the staff were not based on a primary to secondary leak rate of 0.5 gpm per SG. Rather, they were based upon the existing TS limit for primary to secondary leakage of 150 gpd per SG. The staff based their calculations on the 150 gpd per SG because any change to increase the primary to secondary leakage value above 150 gpd would require a TS change. This would entail staff review and approval because the CEA ejection and the seized rotor accidents would require re-analysis of their consequences since their present licensing basis analyses utilize a primary to secondary leak rate of 150 gpd per SG. At that time it would be appropriate to re-assess the consequences of a SGTR accident at the higher primary to secondary leak rate.

The licensee's December 20, 2001, supplemental letter stated that the excerpts from Appendices D and E, provided as attachments to the letter, are not part of the licensing basis for ANO-2 or the license application. The staff disagrees with that. Those appendices, which deal with control room operators' doses, provide a necessary basis for the staff to conclude that the proposed changes are acceptable because the facility meets GDC-19, even with these changes. Therefore, it forms the basis for the staff concluding that GDC-19 is met. Meeting GDC-19 is not just a requirement for the MHA; this requirement covers all accidents.

Table 7.3-4 presents the assumptions utilized by the staff in their confirmatory analysis of the licensee's assessment of a SGTR. The potential dose consequences calculated by the staff are presented in Tables 7.3-5 and 7.3-6. The doses were found to be less than the acceptance criteria noted in SRP Section 15.6.3 and GDC-19 of Appendix A to 10 CFR Part 50, and therefore, are acceptable.

7.3.5 Radiological Analysis Summary

The staff has assessed the impact of the SGTR, the MHA, the FHA, and the CEA ejection accidents to determine whether consequences are within the NRC's acceptance criteria, even with the power uprate and other proposed changes. The staff has concluded that doses would not exceed the guidelines presently contained in the NRC's SRPs for the various accidents and that the facility continues to meet the requirements of 10 CFR Part 100 and GDC-19.

Table 7.3-1 - Assumptions for Maximum Hypothetical Accident Analysis	
Core Thermal Power (MWt)	3087
Activity Released to the Reactor Building	
Airborne (fraction of core)	
Iodine	0.25
Noble Gases	1.0
Iodine Plateout Factor	0
Iodine Species (fraction)	
Elemental	0.91
Particulate	0.05
Organic	0.04
Activity Released to Sump (fraction)	
Iodine	0.5
Noble Gases	0.0
Reactor Building	
Free Volume (ft^3)	1.77E6
Leakage Rate (%/day)	
0-24 hours	0.1
> 24 hours	0.05
Sump Liquid Volume (ft^3)	62,898
Reactor Building Cooling Unit	
Flow Rate (cfm)	5.40E4
Recirculation Efficiency (%)	0
Cooling Start Time after MHA (hr)	0.015
Reactor Building Spray System	
Actuation Time (sec)	63

Table 7.3-1 - Assumptions for Maximum Hypothetical Accident Analysis	
Spray Removal Constants (/hr)	
Elemental	20 for decontamination factor (DF) < 200
	0 for DF ≥ 200
Particulate	3.97 prior to recirculation
	4.24 during recirculation with DF < 50
	0.424 during recirculation with DF ≥ 50
Fraction of Reactor Building Unsprayed	0.224
Recirculation Loop	
Leakage Rate (cc/hr)	2060
Partition Factor	0.1
Minimum Time to Recirculation (hr)	0.43
Passive Component Failure Leak Rate (gpm) for 30 minutes @24 hours post-MHA	50
Partition Factor	0.1
Control Room	
Free Volume (ft^3)	4.0E5
Filtered Recirculation Flow (cfm)	1667
Recirculation Efficiency (%) for all forms of Iodine	95
Makeup Air Filtration Rate (cfm)	333
Makeup Air Filter Efficiency (%) for all forms of Iodine	99
Unfiltered Air Infiltration Rate (cfm)	61
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

Table 7.3-1 - Assumptions for Maximum Hypothetical Accident Analysis	
Atmospheric Dispersion Factors (sec/m^3)	
EAB	6.5E-4
LPZ	
0-8 hours	3.1E-5
8-24 hours	3.6E-6
1-4 days	2.3E-6
4-30 days	1.4E-6
Control Room	
0-2 hours	9.77E-4
2-8 hours	5.76E-4
8-24 hours	2.56E-4
1-4 days	1.68E-4
4- 30 days	1.25E-4
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Table 7.3-2 - Assumptions for Fuel Handling Accidents	
Core Thermal Power (MWT)	3087
Number of Assemblies	177
Highest Power Discharged Assembly	
Axial Peak to Average Ratio	1.7
Radial Peak to Average Ratio	1.7
Occurrence of Accident (hours after shutdown)	100
Damaged fuel rods	60
Activity released from the gap	
Noble gases except ^{85}Kr	0.10
^{85}Kr	0.30
Iodine	0.135
Iodine Gap Inventory	
organic (%)	0.25
inorganic (%)	99.75
Pool DF	
organic (%)	1
inorganic (%)	133
Fuel Handling Building Adsorber Efficiency (%)	0
Atmospheric Dispersion Factors (sec/m^3)	
EAB	6.5E-4
LPZ	
0-8 hours	3.1E-5
Control Room	
0-2 hours	1.2E-3
Breathing Rates (m^3/sec)	
Offsite and Control Room	3.47E-4

Table 7.3-3 - Assumptions for Rod Ejection Accident	
Core Thermal Power (MWT)	3087
Fuel Melt (%)	0
Primary-to-Secondary Leak Rate (gpd per SG)	150
Failed Fuel (% of core fuel)	15
Activity released to reactor coolant from gap (% of core inventory)	
Noble Gases except ^{85}Kr	10
^{85}Kr	30
Iodine	13.5
Iodine Partition Factor in the SGs before and after the accident	0.01
Reactor Building	
Volume (ft^3)	1.78E6
Leak Rate (%/day)	0.1 for t = 0-1 day
	0.05 for t > 1 day
Steam Dump from Relief Valves (lbs per two SGs)	
0-2 hours	Proprietary
2-8 hours	Proprietary
Peaking Factor	1.65

Table 7.3-3 - Assumptions for Rod Ejection Accident	
Atmospheric Dispersion Factors (sec/m^3)	
EAB	6.5E-4
LPZ	
0-8 hours	3.1E-5
8-24 hours	3.6E-6
1-4 days	2.3E-6
4-30 days	1.4E-6
Control Room (Containment Pathway)	
0-2 hours	9.77E-4
2-8 hours	5.76E-4
8-24 hours	2.56E-4
1-4 days	1.68E-4
4- 30 days	1.25E-4
Control Room (Secondary Side Release)	
0-2 hours	6.31E-4
2-8 hours	3.65E-4
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Table 7.3-4 - Assumptions for Steam Generator Tube Rupture Accident	
Iodine Partition Factor for Steaming Releases from Intact SG	0.01
Steam Release from Defective SG	
0-15 minutes	Proprietary
15-30 minutes	Proprietary
30-45 minutes	Proprietary
45-60 minutes	Proprietary
Steam Release from Intact SG (lbs)	
0-1 hour	Proprietary
1-2 hours	Proprietary
2-8 hours	Proprietary
Estimated Break Flow to Faulted SG (lbs)	
0-60 seconds	Proprietary
60 seconds - 30 minutes	Proprietary
30 minutes - 60 minutes	Proprietary
Primary to Secondary Leak	
Maximum for any one SG (gpd)	150
Total for all SGs (gpd)	300
Time to Isolate Faulted SG (sec)	3600
Flashing Fraction	Proprietary
Duration of Plant Cooldown (hrs	8
Breathing Rate	
0-8 hours (m^3/sec)	3.47 E-4

Table 7.3-4 - Assumptions for Steam Generator Tube Rupture Accident	
Primary coolant concentration @ 60 µCi/gm of dose equivalent I-131	Pre-existing Sike Value (µCi/gm)
	I-131 = 41.3
	I-132 = 13.1
	I-133 = 58.4
	I-134 = 6.34
	I-135 = 27.2
Mass of primary and secondary coolant	
Mass of Primary Coolant (lbs)	4.19E5
Mass of Secondary Coolant in SG (lbs)	Proprietary
Primary Coolant Dose Equivalent (DE) I-131 concentration (µCi/gm)	
Maximum Instantaneous Value	60
48 Hour Value	1.0
Secondary Coolant DE I-131 concentration (µCi/gm)	0.10
Primary to Secondary Leak Rate, Total Identified and Unidentified (gpm)	11
Letdown Flow Rate (gpm)	35
Equilibrium Release Rate from Fuel for a Spiking Factor of 500 times the Release Rate for 1.0 µCi/gm of DE I-131	
	I-131 = 3833 Ci/hr
	I-132 = 7445 Ci/hr
	I-133 = 8134 Ci/hr
	I-134 = 8421 Ci/hr
	I-135 = 6807 Ci/hr

Table 7.3-4 - Assumptions for Steam Generator Tube Rupture Accident	
Control Room	
Free Volume (ft^3)	4.0E4
Makeup Filter Efficiency for Iodine (%)	99
Makeup Air Filtration Rate (cfm)	333
Unfiltered Air Infiltration Rate (cfm)	61
Filtered Recirculation Flow (cfm)	1667
Recirculation Flow Filter System Efficiency for Iodine (%)	95
Occupancy Factors	
0-1 day	1.0
Atmospheric Dispersion Factors (sec/m^3)	
Control Room	
0-1 hours	8.05E-4
1-2 hours	6.31E-4
2-8 hours	3.65E-4
EAB	
0-2 hours	6.5E-4
LPZ	
0-8 hours	3.1E-5
Breathing Rate (m^3/sec)	3.47E-4
Spiking Factor for Accident Initiated Spike	500

Table 7.3-5 - Thyroid Doses from Postulated Accidents (Rem)			
Accident	EAB	LPZ	Control Room
1. MHA			
Reactor Building	87	10	12
ECCS Leakage	2.4	1.8	3.0
2. FHA	45	2.2	2.7
3. Rod Ejection			
Reactor Building Release	67	18	22
Secondary Side Release	4.4	2.0	0.81
4. SGTR			
Pre-Existing Spike	39	1.9	1.5
Accident-Initiated Spike	9.2	0.45	0.35

Table 7.3-6 - Whole Body Doses from Postulated Accidents (Rem)			
Accident	EAB	LPZ	Control Room
1. MHA			
Containment	3.2	0.31	0.15
ECCS Leakage	6.9E-3	1.7E-3	4.9E-5
2. FHA	8.8E-2	4.2E-3	5.1E-3
3. Rod Ejection			
Reactor Building Release	0.27	0.032	4.4E-3
Secondary Side Release	0.39	0.041	2.2E-3
4. SGTR			
Pre-Existing Spike	0.04	1.9E-3	4.4E-5
Accident-Initiated Spike	0.033	1.6E-3	3.4E-5

8.0 PROBABILISTIC RISK ASSESSMENT (PRA)

To evaluate the impact on risk at ANO-2 from the proposed extended power uprate, the licensee assessed their plant-specific PRA for internal events and their external events analyses. The staff reviewed the information provided in the licensee's application dated December 19, 2000, specifically Section 9.12 of Enclosure 5, and the licensee's associated supplemental information and responses to the staff's RAIs dated June 28, July 24, October 12, November 16, and December 20, 2001. In addition, in December 2001, the NRC staff conducted a site review of the ANO-2 PRA, specifically in the areas of their fire analysis and human reliability analysis, to support its review of the licensee's proposed extended power uprate.

The staff used the guidance provided in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to focus the review of this non-risk informed submittal. The staff's evaluation of the licensee's application focused on the capability of the licensee's PRA to analyze the risks stemming from both current, pre-extended power uprate plant operation and the proposed post-extended power uprate conditions. The staff's evaluation did not involve an in-depth review of the licensee's PRA, but rather focused on whether the extended power uprate would raise questions regarding the licensee's ability to provide adequate protection by meeting the deterministic requirements and regulations. This evaluation included a review of the licensee's discussions of extended power uprate impacts on core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events, and shutdown operations. The evaluation also addressed the quality of the ANO-2 PRA, commensurate with its use in the licensee's and staff's decision-making processes.

8.1 Internal Events

The NRC staff evaluation report on the ANO-2 individual plant examination (IPE) was issued in May 1997 and concluded that the licensee had met the intent of GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." The licensee has updated the PRA several times since the staff review relative to GL 88-20 to maintain it consistent with the as-built plant.

The licensee evaluated the changes due to extended power uprate implementation for its potential impact on the PRA models for internal events in the following key areas: initiating event frequency, component reliability, system success criteria, and operator response. Each of these areas are specifically addressed in the following subsections, with the staff's evaluation findings, followed by a description of the overall impacts on CDF and LERF from internal events for the extended power uprate.

8.1.1 Initiating Event Frequency

The ANO-2 internal events PRA addresses LOCAs, SGTRs, LOOP transients, spurious main steam isolation signal (MSIS), loss of support systems, and ATWS.

For LOCAs, which include small-, medium-, and large-break sizes, the licensee stated that the frequency is dictated by the potential for passive pipe failures. Since these failure frequencies are independent of the power level, the LOCA frequencies are not affected by the extended power uprate.

For SGTR events, the licensee considered the fact that RSGs were installed in the last refueling outage (i.e., 2R14). The RSGs are designed with an increased tubing surface area to accommodate the extended power uprate and the tubing material is Alloy 690, which was chosen as the optimum material for RSGs due to its significantly improved corrosion resistence when compared to Alloy 600. In addition, other enhancements were made in the RSG design to provide margin against degradation. These enhancements include improved tube-to-tube joint design, anti-vibration bar design, and tube support plate design. Based on these considerations, the licensee concluded that the currently modeled frequency of SGTRs, which is based on the industry experience of tube ruptures, is applicable to extended power uprate conditions.

The licensee stated that the LOOP frequency is dictated by the reliability of the switchyard and grid, which are not degraded by the extended power uprate. The staff has reviewed the licensee's evaluation of ANO-2 grid stability and related electrical equipment (e.g., transformers) and determined that they are acceptable for extended power uprate conditions. This evaluation included the determination of acceptability of the main transformer without the installation of additional cooling, which was originally planned by the licensee and identified in its original submittal to be installed in the next refueling outage, but has since been deferred to a later outage. Therefore, even though the main transformer may be operated slightly beyond its current rating for extended power uprate conditions without the installation of additional cooling, the staff has determined that this condition is acceptable and will not degrade the performance of the transformer, as discussed in Section 5.18.3 above.

For transient events, the licensee assessed the trip setpoints for the RPS, feedwater control system, pressurizer level control system, pressurizer pressure control system, and steam dump and bypass control system. As appropriate, setpoints were adjusted to ensure proper plant response to certain equipment failures (e.g., heater drain pump or condensate pump failure) and plant maneuvers (e.g., load changes, heatup, and cooldown). The licensee's original application identified hardware changes and setpoint modifications that were made to maintain margin on trip setpoints at extended power uprate levels and maintain equipment operation within design constraints. These changes include: improved stator water cooling supply, rewound main generator, replaced high-pressure turbine assembly and four stages of low-pressure turbine assembly, improved feedwater system reactor trip override and system response, modified heater drain pumps, and increased condenser surface area. The licensee concluded that the plant modifications and setpoint changes should mitigate the potential for any increase in the frequency of transient events and therefore, the extended power uprate will not increase the frequency of these transient initiators.

A spurious MSIS was evaluated by the licensee in the ANO-2 PRA as a unique initiating event. A spurious MSIS signal will close the MSIVs and isolate component cooling water and auxiliary cooling water, which will lead to the loss of the main feedwater and condensate pumps. Closure of the MSIVs will also result in a loss of steam flow, challenge the MSSVs, challenge the primary safety relief valves, and render the turbine bypass valves unavailable. Recently, the licensee also added the containment spray actuation signal (CSAS) on high-high containment building pressure to terminate main feedwater and main steam flow from an unaffected RSG to ensure the isolation of these systems following a MSLB. This modification was added as part of the RSG effort. The CSAS provides a parallel actuation of selected components actuated by the MSIS and thus, introduces another possible spurious signal path. However, concurrent with the addition of the CSAS, the MSIS and CSAS logic circuits were triphardened, requiring at least two relay failures to isolate the main feedwater or main steam system. In the prior MSIS signal logic, an individual relay failure could cause a plant trip and loss of feedwater. This modification reduces both the frequency of a spurious signal initiator and the probability of a spurious signal after a plant trip. Thus, the licensee concluded that the net effect of the two changes (i.e., addition of CSAS, and trip-hardening CSAS and MSIS logic circuits) is expected to actually reduce the CDF contribution due to a spurious signal.

For support systems that may initiate a transient, such as AC power, direct current (DC) power, and SWS, the licensee indicated that none are adversely affected by the extended power uprate. The licensee further stated that these systems were being reviewed for any changes in loads and requirements and verified to be within the design capacities of the systems.

For ATWS events, the licensee stated that the sequence initiation would occur under the same primary system conditions both before and after extended power uprate. Since there is no change in the transient initiator frequency, as identified above, which is an input to calculating the ATWS event frequency, the licensee concluded that the potential for an ATWS event is also unchanged.

The licensee concluded that the extended power uprate will have no adverse affect on the internal events PRA initiator frequencies and any future deviations in initiating event frequencies will be identified by existing monitoring processes, such as licensee event reports, condition reporting, and industry events databases. In addition, safety system actuations are trended under the maintenance rule as an indicator of unnecessary challenges of safety-related equipment.

The staff finds that it is reasonable to conclude that the initiating event frequencies will not change, as long as the operating ranges or limits of equipment are not exceeded. In addition, the staff notes that if there are any changes observed in the future in initiating event frequencies, these changes will be identified and tracked under the plant’s existing performance monitoring programs and processes.

The licensee's risk evaluation for the extended power uprate did not explicitly model, for the loss of feedwater initiating event, the impact of a spurious MSIS or CSAS actuation that is created by the signal logic additions and modifications. Though not explicitly addressed in the extended power uprate PRA modeling of the loss of feedwater initiating event, the licensee has stated that they have entered the signal logic modification into the ANO-2 PRA model change request process to ensure that it is considered for inclusion in a future revision of the ANO-2 PRA model. In addition, the licensee's risk evaluation for the extended power uprate did not model the potential for an increase in the frequency of a main transformer failure due to this transformer being operated beyond its current rating under extended power uprate conditions. However, given that the staff’s expectation is that the trip-hardening of the MSIS and CSAS actuation logics will actually decrease the potential for an inadvertent loss of feedwater event and the staff has determined that the main transformer’s reliability should not be degraded under extended power uprate conditions, the staff believes that further resolution of these issues would not significantly alter the overall results or conclusions for this license amendment. Therefore, the staff concludes that these issues would not warrant denial of this license amendment and that the expectation is that there will be no change in initiating event frequencies as a result of the extended power uprate.

8.1.2 Component Reliability

The licensee stated that they performed comprehensive reviews of all plant systems and associated equipment with the potential to be affected by the extended power uprate, and that any changes to equipment service conditions and processes were identified in their extended power uprate submittal. The licensee stated that modifications were made to improve the performance of certain equipment and systems under extended power uprate conditions to ensure plant systems and equipment will continue to be operated within design constraints and that component failure rates will not change with the implementation of the extended power uprate.

The licensee indicated that they will rely on existing component monitoring programs, such as preventive maintenance, vibration analysis, thermography, oil analysis, EQ, erosion/corrosion, and maintenance rule to identify any additional wear as a result of the extended power uprate. While the extended power uprate may result in some components being refurbished or replaced more frequently, the licensee asserted that the functionality and reliability of components will be maintained to the current standard. These monitoring programs are also expected to identify any future deviations in component failure rates. Further, the licensee indicated that train-level changes to equipment unavailability for systems modeled in the ANO-2 PRA are tracked as part of their PRA model maintenance process and that they periodically review equipment unavailabilities and update the model, as appropriate.

The licensee concluded that the extended power uprate will have no adverse affect on component failure rates and that existing monitoring programs will be used to identify any future deviations in component failure rates.

The staff finds that it is reasonable to conclude that equipment reliability will not change, as long as the operating ranges or limits of the equipment are not exceeded. For equipment that is operated within its operating ranges or limits, the staff notes that the licensee's component monitoring programs, as identified above, should detect any significant degradation in performance and the staff expects these programs to maintain the current reliability of the equipment.

As stated in the prior subsection on initiating event frequencies, the licensee's risk evaluation for the extended power uprate did not model the potential for an increase in the frequency of a main transformer failure due to this transformer being operated beyond its current rating under extended power uprate conditions. However, given that the staff has determined that the main transformer's reliability should not be degraded under extended power uprate conditions, the staff believes that further resolution of this issue would not significantly alter the overall results or conclusions for this license amendment. Therefore, the staff concludes that this issue would not warrant denial of this license amendment and that the expectation is that there will be no change in component reliability as a result of the extended power uprate.

8.1.3 Success Criteria

The licensee stated that they performed a detailed review to identify the effect of the extended power uprate on the system success criteria credited in the ANO-2 internal events PRA model. The success criteria specify the requirements of the plant systems to address critical safety functions. These critical safety functions include: reactivity control, RCS pressure control/pressure boundary integrity, RCS and core heat removal, once-through-cooling, RCS inventory control, and long-term RCS inventory control and heat removal.

With one exception, the extended power uprate was determined by the licensee to have no impact on the success criteria associated with each of the critical safety functions. The one exception is that the high pressure safety injection (HPSI) success criteria for long-term RCS heat removal following a LBLOCA was increased from one HPSI pump injecting into the RCS through two of four open injection flow lines (one of which may contain a pipe break) for the current power level, to one HPSI pump injecting into the RCS through three of four open injection flow lines (one of which may contain a pipe break) for the extended power uprate conditions. The difference in success criteria is due to the slightly higher decay heat level under extended power uprate conditions at the time of initiation of recirculation.

The staff finds that the extended power uprate will result in a change to the success criteria for the LBLOCA HPSI recirculation mode. The impact of this change, along with the other impacts from the internal events evaluation on CDF and LERF is presented in Section 8.1.5.

8.1.4 Operator Response

The licensee stated that they performed a detailed assessment to determine the effects of the proposed extended power uprate on operator actions and their associated human error probabilities (HEPs). The extended power uprate tends to impact operator actions by reducing the time available for the operator to complete recovery actions. This change is due to the higher decay heat level at extended power uprate conditions. The licensee did not identify any new risk-significant operator actions as a result of the extended power uprate.

The licensee's analysis involved the development of available times for post-initiator operator actions using the ANO-2 CENTS model. The licensee described the CENTS model as a best-estimate deterministic thermal-hydraulic analysis code that accounts for both the primary and secondary system response to plant transients, including SBLOCAs. The ANO-2 CENTS model was exercised to estimate the latest time for successful operator action to avert two-phase core uncovery for a representative set of accident scenarios that the PRA model identified as important to risk. These CENTS-generated core uncovery times were then used as the pre- and post-extended power uprate available times for operator actions. The resulting available times were then input into the ANO-2 PRA human reliability analysis model to determine the appropriate HEPs for each operator action. Since core uncovery precedes core damage, the licensee stated that the use of core uncovery time as the basis for the available time for operator actions yields conservatively high HEPs. Thus, the HEPs applied in the assessment of the risk impact of the ANO-2 extended power uprate are considered by the licensee to be conservatively high.

The licensee stated that their human reliability analysis involved a compilation of state-of-the-art methods. Pre-initiator human errors were quantified using a simplified form of the technique for human error rate prediction that was developed for the accident sequence evaluation program. Proceduralized post-initiator human errors were quantified using two complementary approaches: 1) the human cognitive reliability correlation developed by EPRI, which incorporates data from the operator reliability experiments that are described in EPRI Report NP-6560L, "A Human Reliability Analysis Approach Using Measurements for Individual Plant Examinations," and Topical Report TR-100259, "An Approach to the Analysis of Operator

Actions in Probabilistic Risk Assessment," and 2) the cause-based methodology developed by EPRI, which is also documented in EPRI Topical Report TR-100259. Non-proceduralized post-initiator human errors were quantified using a revised systematic human action reliability procedure that was developed by EPRI and documented in EPRI Topical Report TR-101711, Tier 2, "A Revised Systematic Human Action Reliability Procedure." The licensee stated that dependencies between post-initiator operator actions were accounted for using the revised systematic human action reliability procedure method developed by EPRI in TR-101711, Tier 2. Finally, the licensee stated that, aside from the fire analysis which is discussed below in Section 8.2.2, whenever more than one operator action occurs in a single cutset, a single combined operator action is applied that effectively replaces the contribution from the individual actions and the HEP value of the combined operator action accounts for the dependencies between the individual actions.

The licensee provided a list of the post-initiator operator actions whose HEP value changed as a result of the extended power uprate conditions. This list is provided below in Table 8.1, including the event name, event description, and pre- and post-extended power uprate available times and HEPs. The available times identified in Table 8.1 correspond to the system time window of the human cognitive reliability method, which includes the manipulation/execution time.

Table 8.1 - Post-Initiator Operator Actions Affected by Extended Power Uprate

Event Name	Description	Pre-Extended Power Uprate		Post-Extended Power Uprate	
		Available Time (minutes)	HEP	Available Time (minutes)	HEP
EHF2A1A2SP	Failure to re-energize 2A1/2A2 from ST2 (SBLOCA or SGTR)	42	1.9E-1	39	2.9E-1
EHF2A1A2TP	Failure to re-energize 2A1/2A2 from ST2 (Transient)	80	1.6E-2	68	2.0E-2
EHF2A3A4XP	Failure to reduce loads and cross-tie 4160V buses 2A3 and 2A4	80	1.1E-2	68	2.8E-2
EHF2AAC17P	Failure to open manual valve 2AAC-17	80	4.5E-4	68	1.2E-3
EHF2AACSTP	Failure to start and align AAC generator following failure of an emergency DG to start	80	2.6E-3	68	5.6E-3
EHF2B5B6XP	Failure to cross-tie 480V vital buses 2B5 and 2B6	80	9.5E-3	68	1.7E-2
EHF2LOSP2P	Failure to align offsite power to 2A1/2A2	80	1.3E-2	68	1.4E-2
EHF2LOSPCP	Failure to align offsite power to 2A1/2A2	42	8.4E-2	39	1.2E-1
EHF2LOSPXP	Failure to align offsite power to 2A1/2A2	80	4.3E-3	68	6.0E-3
EHF2Y1Y2XP	Failure to cross-tie 120V AC vital buses 2Y1 and 2Y2	80	2.9E-3	68	6.7E-3
FHF2FWST3P	Failure to recover main feedwater following loss of SUT3	80	6.0E-3	68	6.3E-3
FHF2MFWTRP	Failure to restart tripped main feedwater pump	80	1.9E-3	68	4.1E-3
PHF2MSSVGP	Failure to maintain ruptured RSG pressure less than MSSV setpoint	120	1.2E-3	112	1.6E-3
QHF22P75SP	Failure to establish flow to RSGs from auxiliary feedwater pump	80	2.9E-4	68	5.8E-4
QHF2A1CSRP	Align emergency feedwater/auxiliary feedwater suction source to the qualified condensate storage tank (T-41B) (SGTR)	42	3.2E-1	39	5.0E-1
QHF2A1CSTP	Align emergency feedwater/auxiliary feedwater suction source to the qualified condensate storage tank (T-41B) (Transient)	80	3.3E-3	68	1.0E-2
QHF2A1CSXP	Align emergency feedwater/auxiliary feedwater suction source to the qualified condensate storage tank (T-41B) (Long-term)	122	3.7E-3	113	6.5E-3
THF2OTCLRP	Failure to establish once-through cooling after total loss of feedwater during SGTR	52	4.7E-2	42	1.2E-1
THF2OTCLTP	Failure to establish once-through cooling after total loss of feedwater	80	1.3E-3	68	3.8E-3

In addition to the above information, the licensee provided a list of the dominant (i.e., top fifteen) operator actions, based on their Fussell-Vesely (F-V) importance to CDF value. This list is provided below in Table 8.2. Many of the operator actions identified in Table 8.1 are also identified as being important, based on their F-V importance values. Table 8.2 presents the dominant operator actions along with their pre- and post-extended power uprate calculated F-V importance values.

Table 8.2 - Post-Initiator Operator Actions Affected by Extended Power Uprate

Event Name	Description	Pre-Extended Power Uprate F-V Importance	Post-Extended Power Uprate F-V Importance	Change in F-V Importance (%)
QHF2EFWBXX	Failure to manually control emergency feedwater pump (2P7B) speed and discharge valves	5.11E-1	4.63E-1	-9%
DHF2D31BAP	Operator fails to transfer 2D31B to alternate power source	5.02E-1	4.39E-1	-13%
QHF2EFWAXX	Failure to manually control emergency feedwater pump (2P7A) speed and discharge valves	4.94E-1	4.33E-1	-12%
EHF2LOSPXP	Failure to align offsite power to 2A1 or 2A2 after auto realign fail	9.32E-2	9.91E-2	6%
EHF2B5B6XP	Failure to cross-tie 480V vital buses 2B5 and 2B6	8.20E-2	8.81E-2	7%
DHF2D32BAP	Operator fails to transfer 2D32B to alternate power source	7.00E-2	6.28E-2	-10%
THF2OTCLTP	Failure to establish once-through cooling after total loss of feedwater (Transient)	6.45E-2	1.18E-1	83%
EHF2AACSTP	Failure to start and align AAC generator following failure of an emergency DG to start	5.84E-2	6.42E-2	10%
EHF2A3A4XP	Failure to reduce loads and cross-tie 4160V buses 2A3 and 2A4	4.85E-2	5.36E-2	11%
EHF2A1A2TP	Failure to re-energize 2A1/2A2 from ST2 (Transient)	4.61E-2	4.73E-2	3%
QHF2QCSTXP	Failure to align emergency feedwater/auxiliary feedwater suction source to the qualified condensate storage tank on lo-lo level in 2T-41A/B	4.05E-2	3.53E-2	-13%
QHF22P75SP	Failure to establish flow to RSGs from auxiliary feedwater	3.21E-2	4.34E-2	35%
THF2RCPTRP	Failure to trip RCPs after loss of component cooling water	2.88E-2	2.70E-2	-6%
FHF2SGOFMP	Failure to control main feedwater to prevent high RSG level after malfunction	2.65E-2	3.12E-2	18%
DHF2D31BCP	Failure to align DC bus 2D01 to alternate charger 2D31B	2.12E-2	< 2.70E-2	unknown (probably neg.)
THF2OTCLSP	Failure to establish once-through cooling after total loss of feedwater (SBLOCA)	< 2.12E-2	6.23E-2	unknown (at least 194%)

During the staff site review, the human reliability analysis methodology implemented by ANO-2 and selected HEP calculations were reviewed. One result of the staff site review was that another operator action was identified as being impacted by the extended power uprate, though it had not been previously identified by the licensee. This operator action (THF2OTCLSP) is essentially identical to the operator action that involves a SGTR (operator action THF2OTCLRP), which was identified as being impacted, except that it involves a SBLOCA. It was determined by the staff that even though the operator action was not identified in the list of affected operator actions, its HEP values were revised appropriately in the pre- and post-extended power uprate PRA.

Further, the staff observed that the importance of most operator actions change only slightly when comparing the pre- and post-extended power uprate F-V importance values. Of the fifteen operator actions identified as most important for pre-extended power uprate conditions, fourteen are also identified as most important for post-extended power uprate conditions. Only operator action DHF2D31BCP is dropped from the post-extended power uprate top fifteen in F-V importance list, being replaced by operator action THF2OTCLSP. The staff did note that the operator action of establishing once-through cooling after a loss of all feedwater is increased substantially in importance for both the transient and SBLOCA conditions (operator actions THF2OTCLTP and THF2OTCLSP, respectively). The timing associated with performing these operator actions are affected by the increased decay heat levels at extended power uprate. There is also a substantial increase in importance for the operator action to establish auxiliary feedwater to the RSGs (operator action QHF22P75SP). This operator action is impacted by the previous RSG modification involving the CSAS that is described above in Section 8.1.1. No other dominant operator action increases or decreases in importance by more than 20%.

Finally, the staff reviewed a selection of other operator actions (e.g., EHF2LOSPXP and QHF2A1CSTP) to evaluate how closely the licensee's approach adhered to the cited EPRI methodologies, including the consideration of multiple recoveries and dependencies. The staff found for the selected operator actions that the licensee was consistent with the EPRI methodologies.

The licensee has used a compilation of human reliability analysis methodologies to calculate the operator action HEPs for the pre-extended power uprate and post-extended power uprate plant conditions. The differing HEP values for the same operator action reflect the reduction in available time to diagnose and perform these actions, due to the increased decay heat levels caused by the extended power uprate. These human reliability analysis methodologies have not been formally reviewed and approved by the NRC. However, these methodologies are accepted by a relatively large number of licensee PRA staff and by some PRA consultants, who use them to provide a means to estimate HEP values in a relatively coherent way that recognizes the influence of some situational characteristics (e.g., available response time). The staff recognizes that the identified small changes in the available operator action response times due to the extended power uprate are insignificant in relationship to the uncertainties in estimating their HEP values. As a result, the estimated absolute values for the HEPs cannot be used as the sole basis for determining the acceptability of a license application. However, the evaluations can provide insights into the relative importance, or change in importance, of selected operator actions and can be used to focus the staff review of the license application on those aspects impacted by the extended power uprate.

The staff finds, based on the information provided by the licensee and the staff site review, that the licensee's human reliability analysis application is consistent with the identified methodologies and that the assumed increases in the HEP values for the identified operator actions reasonably reflect the reductions in the times available for the operators to perform the necessary actions under the extended power uprate conditions. The impact on CDF and LERF from these reductions in operator response times is presented below in Section 8.1.5.

8.1.5 Summary of Internal Events Evaluation Results

The licensee indicated that no impacts are expected due to the extended power uprate for initiating event frequencies or component reliability, but potential impacts of the extended power uprate were identified for the success criteria of HPSI in the recirculation mode for LBLOCA events and for selected operator actions due to the decrease in available operator response times.

The majority of the change from the pre-extended power uprate to the post-extended power uprate CDF occurs in TBF sequences, which increase in CDF by about 9%. The TBF sequence involves transient initiating events (T) with a subsequent loss of RCS and core heat removal (B) and failure of once-through cooling (F). This sequence also leads to a high RCS pressure with early core damage. A similar sequence, TQBF, accounts for almost another 5%. The TQBF sequence is the same as the TBF sequence with the additional loss of RCS pressure control/pressure boundary integrity (Q), such as from a stuck-open safety relief valve. Lesser impacts are from RBF and SBF sequences, which are similar to the TBF sequence except that the initiating event is a SGTR (R) and a SBLOCA (S), respectively. In nearly all cases, the increase in CDF is from sequences that involve the loss of RCS and core heat removal, and failure of once-through cooling. Core heat removal failure is the result of failures of main, emergency, and auxiliary feedwater. Failure of once-through cooling is due to HPSI or ECCS failures and failures of the LTOP vent valve. These impacts result in about a 16% increase in internal events CDF to almost 2.0E-5/year. This represents an increase of about 2.7E-6/year from the current CDF of approximately 1.7E-5/year.

As with other large, dry containment plants, the LERF for ANO-2 is dominated by bypass events, such as SGTR and interfacing system LOCAs, along with transients that involve failure of primary-to-secondary cooling and once-through cooling. From the pre-extended power uprate LERF results, the potential for a LERF was estimated to be about 3.9E-7/year. This value is small; approximately 2% of the total pre-extended power uprate CDF. The post-extended power uprate LERF was calculated by the licensee to be about 4.8E-7/year, an increase of about 9.3E-8/year, or an increase of about 24%. Similar to the pre-extended power uprate case, the post-extended power uprate LERF value is also roughly 2% of the post-extended power uprate CDF. Virtually all of the changes in the LERF quantification are a result of the changes made to the Level 1 accident sequence analysis in the form of reduced times available for operator actions in preventing core damage. The increase in the LERF value is dominated by SGTR events; specifically sequence RBF.

Based on the reported analyses and results, the staff finds that the changes in CDF and LERF from internal events due to the proposed extended power uprate are small and very small, respectively, and are within the acceptance guidelines provided in RG 1.174.

8.2 External Events

The NRC's staff evaluation report on the ANO-2 IPE of external events (IPEEE) was completed in February 2001 and it concluded, based on the staff's screening review, that the licensee's process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and therefore, that ANO-2 had met the intent of Supplement 4 to GL 88-20.

The licensee evaluated the changes due to extended power uprate implementation for its potential impact on the external events analyses, specifically seismic events, fires, and high winds, floods, and other external events. Each of these external events are individually addressed in the following subsections, followed by the staff's findings regarding the impact of extended power uprate on external events.

8.2.1 Seismic Events

For the IPEEE seismic analysis, ANO-2 is categorized as a 0.3g focused-scope plant, per NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." The licensee performed the ANO-2 seismic evaluation using the EPRI seismic margins assessment methodology described in EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin." Therefore, the licensee did not quantify a seismic CDF. Based on the NRC staff evaluation report on the ANO-2 IPEEE, the licensee stated that the overall high confidence of a low probability of failure plant capacity would be equal to or greater than the review level earthquake value of 0.3g after the resolution of certain outliers. In response to a staff RAI, the licensee stated that all outliers for ANO-2 have been resolved in accordance with their proposed resolutions, as presented in their IPEEE, and that all of their relied-upon equipment meets a minimum review level earthquake of 0.3g. Further, the licensee stated in their supplemental information for the extended power uprate license amendment that the extended power uprate does not modify the safe shutdown pathway assumed in the IPEEE seismic margins assessment. In addition, the licensee stated in their supplemental submittal that the RCS seismic analysis determined the dynamic response of the RCS with RSGs, considering system operating parameters consistent with the extended power uprate, and found the original analysis to remain bounding for the RSG configuration. Also, the licensee relies upon their existing component monitoring programs to account for any additional wear that might result from the extended power uprate and has stated that they will maintain the functionality and reliability of components to the current standard. Therefore, the power level is not expected to affect equipment survivability nor equipment response during an earthquake. Thus, the licensee concluded that the IPEEE seismic margins assessment is not impacted by the extended power uprate and that there will be only a negligible adverse impact from seismic events at extended power uprate conditions.

8.2.2 Fires

For the IPEEE fire analysis, the licensee's analysis is based on the EPRI fire-induced vulnerability evaluation methodology, as described in EPRI Technical Report TR-100370, with some model parameters and analysis guidance taken from the EPRI fire risk analysis implementation guide, as described in EPRI Report 3385-01. As stated in EPRI TR-100370, the fire-induced vulnerability evaluation methodology is oriented toward uncovering limiting plant design or operating characteristics (i.e., vulnerabilities) that make certain fire-initiated events more likely than others.

The licensee estimated in its original IPEEE submittal that the contribution to CDF from six unscreened fire zones was about 3.8E-5/year for ANO-2. During the NRC staff review of the IPEEE, the licensee reevaluated certain fire scenarios and assumptions. As a result, several fire zones were assigned CDFs that increased by about a factor of two from the original IPEEE submittal. The reanalysis resulted in fifteen zones being above the screening criteria (i.e., greater than 1.0E-6/year) and an estimated fire CDF of about 9.5E-5/year, with the largest contribution from a turbine building fire estimated as almost 3.7E-5/year. Though the estimated CDF value more than doubled because of the changes, the licensee stated that they were consistent with the guidance of Nuclear Management and Resources Council (NUMARC) 91-04, Revision 1, "Severe Accident Issue Closure Guidelines," in that the changes were not significant enough to require procedure changes or plant modifications and closure was obtained individually on each significant fire compartment. For the unscreened fire zones, NUMARC 91-04, Revision 1, guidance, in part, requires licensees to find a cost-effective treatment in EOPs or other plant procedures or, if that could not be achieved, to ensure severe accident management guidelines are in place. For the fifteen unscreened fire zones at ANO-2, EOPs and AOPs are in place to address the loss of any systems due to a fire.

The effect of the extended power uprate on the BOP fire analysis is to increase the overall fire CDF by about 1.6E-5/year, an increase of about 17%, to an extended power uprate CDF of approximately 1.1E-4/year. This increase is a result, in part, of the high fire CDF estimates caused by the licensee's decision to discontinue further detailed analyses once the severe accident closure guidelines criteria were satisfied. The licensee attributed the large CDF and CDF increase to:

1) The fire initiator frequencies of the zones conservatively include some components that do not represent credible ignition sources. This increases the absolute value of the fire zone CDF value.

2) The conditional core damage probability is affected by conservatively assuming that a fire damages all equipment located within a given compartment, with the exception of those components that were determined not to fail by a specific fire model. Fire modeling techniques were only utilized on a small percentage of the total number of components located within a given fire compartment. This increase in the conditional core damage probability further magnifies the conservative fire frequency, noted above, and subsequently increases the absolute value of the fire zone CDF value.

3) Multiple operator recovery actions are applied to individual core damage cutsets. This tends to increase the change in the CDF between the pre- and post-extended power uprate CDF results, because more than one operator recovery action HEP may be increased in each cutset.

The licensee identified as an additional conservatism the use of available times for operator recovery actions that are based on the CENTS-generated time to core uncovery as opposed to the time to core melt. In the licensee's supplemental letter dated June 28, 2001, the licensee stated that the time to core uncovery after all core cooling was lost, with heat removal through the MSSVs, was 80 minutes pre-extended power uprate and 68 mintues post-extended power uprate. Because fires are neither expected to cause LOCAs (because fires don't cause pipes to rupture and because the IPE models the Byron Jackson RCP seals as unlikely to fail if the RCPs are tripped within 30 minutes), nor fail MSSVs, most of the recoveries applied in the fire analysis use this time to core uncovery as the available time for operator recovery actions. The fire analysis was based on the original IPE results with the equipment assumed failed assigned a failed state. This may introduce another conservatism into the fire analysis because the original IPE had a CDF that is about a factor of two greater than that of the current internal events PRA.

The dominant fire zones under extended power uprate conditions do not change from those that dominate the pre-extended power uprate fire analysis results. Fires in the turbine building, cable spreading room, intake structure, diesel corridor, and electrical equipment room make up almost 80% of the fire CDF in the pre- and post-extended power uprate fire analysis. Likewise, the increase in fire CDF is dominated by fires in the cable spreading room, diesel corridor, electrical equipment room, lower south electrical/piping penetration room, and south switchgear room. These five fire zones make up over 72% of the increase in fire CDF. The licensee did not include the previously screened fire zones in the change in risk analysis because several unscreened zones were more than a factor of ten greater than the 1E-6/year screening values. Changes on the order of a factor of ten were not expected, given that the only changes in the pre- and post-extended power uprate fire analysis were changes in the time available for operator recovery actions.

The screening methodology applied by the licensee makes less and less conservative assumptions until a fire zone is screened out, the results do not indicate a vulnerability, or a vulnerability is identified and addressed. If applied correctly, this type of analysis will always produce a conservative result. None of the screened fire zones and only two of the unscreened zones were divided into smaller sub-zones. The two divided unscreened zones were only divided into two sub-zones. Therefore, there is a large amount of equipment assumed to be potentially failed in each zone. A further step in the screening process is to examine the physical characteristics of some zones with high estimated CDFs and to apply fire modeling techniques to identify specific equipment that would not be expected to be failed by a credible fire in the zone. In the detailed documentation examined at the site, it was noted that the long list of equipment that could possibly be failed in a fire zone was followed by a list of equipment determined not to have failed based on the fire modeling techniques. It was observed that this second list usually included only one or two components, and no second list contained more than three components. This indicates that the uncertain fire initiation and propagation methodologies were not excessively applied. Therefore, the staff finds that the quantitative results are, with reasonable assurance, a conservative estimate of the fire CDFs for each zone. Although the subsequent use of multiple independent recovery actions in cutsets may not be conservative, multiple excessively small HEPs would not have yielded the relatively high CDFs. This conclusion is supported by the relatively high estimate of the total fire CDF when compared to other plants.

The licensee applied multiple, independent recovery operations to the cutsets instead of an integrated recovery action specific to each cutset. The licensee reviewed these recovery actions and, in some cases, replaced the independent recovery actions with a single dependent action. During the site review, cutsets with two or three independent recovery actions were observed, as well as other cutsets where independent recovery events had been replaced with a dependent event. The licensee’s HEPs methodology includes time-dependent diagnosis and decision-making in each action. Each independent action will include the time for diagnosis and decision-making, whereas it can reasonably be argued that several dependent actions might logically follow from one diagnosis and decision-making activity. Although the use of the independent actions may result in a smaller total HEP for each cutset, the staff finds that the total change due to the change in available time will most likely be calculated as larger for multiple independent actions than for a single dependent action. The licensee stated that the only difference in the pre- and post-extended power uprate fire analysis was a change in the time available for operator recovery actions, and that the dominant recovery time change was a decrease from 80 minutes to 68 minutes. This is a relatively small change that, more importantly, still provides the licensee's personnel 68 minutes before core uncovery; a length of time that should allow for attempts to recover equipment. Therefore, the staff finds that there is reasonable assurance that the change estimated in the fire CDF is conservative because the baseline CDF estimates are conservative and the use of the multiple independent recovery actions in most cutsets amplifies the impact of the change in the recovery time by including diagnosis and recovery independently in each recovery action.

The extended power uprate fire analysis is slightly above the acceptance guidelines of RG 1.174. The licensee's risk evaluation for the extended power uprate did not include additional confirmatory analyses of the fire zones to ensure that, without the conservatisms used in the current analysis and using the most recent model (i.e., PRA Revision 2.1 instead of the IPE), the results would indeed result in only a small increase in risk and be within the acceptance guidelines presented in RG 1.174. However, the staff believes that further resolution of this issue would not significantly alter the results or overall conclusions of this specific license application. Therefore, the staff concludes that this issue would not warrant denial of this license amendment.

8.2.3 High Winds, Floods, and Other External Events

For the IPEEE evaluation of high winds, floods, and other external events, the licensee used the progressive screening approach described in NUREG-1407. The licensee did not quantitatively estimate the CDF contribution from high winds, floods, and other events since these events were screened out using their progressive screening approach. The licensee did address in their IPEEE the aspects of their plant design that do not conform to the criteria provided in the 1975 SRP, due to the age of the plant, and indicated that the extended power uprate conditions do not change the results of this evaluation, which were that the impacts of these non-conformances do not exceed the NUREG-1407 screening criteria (i.e., 1.0E-6/year). Therefore, the licensee concluded that the extended power uprate has only a negligible adverse affect on these other external events.

8.2.4 Staff Findings Regarding Impacts of Extended Power Uprate on External Events Analyses

Given that the licensee has resolved all previously identified seismic outliers for ANO-2 in accordance with their proposed resolutions, as presented in their IPEEE, that all of their relied-upon equipment meets a minimum review level earthquake of 0.3g, and that the extended power uprate does not create any new seismic vulnerabilities, the staff finds that the increase in CDF from seismic events due to the proposed extended power uprate is negligibly small and within the acceptance guidelines provided in RG 1.174. Further, for the high winds, floods, and other external events, the licensee has previously addressed those aspects of their plant design that do not conform to the criteria provided in the 1975 SRP and, based on the fact that the extended power uprate does not impact these high winds, floods, and other external events analyses, the staff finds that the increase in CDF from high winds, floods, and other external events due to the proposed extended power uprate is also negligibly small and within the acceptance guidelines provided in RG 1.174.

As stated above in Section 8.2.2, the extended power uprate fire analysis results are slightly above the acceptance guidelines of RG 1.174. The licensee's risk evaluation for the extended power uprate did not include additional confirmatory analyses of the fire zones to ensure that, without the conservatisms used in the current analysis and using the most recent model (i.e., PRA Revision 2.1 instead of the IPE), the results would indeed result in only a small increase in risk and be within the acceptance guidelines presented in RG 1.174. However, the staff believes that further resolution of this issue would not significantly alter the results or overall conclusions of this specific license application and thus, would not warrant denial of this license amendment.

8.3 Shutdown Risk

The licensee indicated that they examined shutdown risks in a qualitative manner by addressing the questions posed in Table III-1 of SRP 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," to determine if the impacts on shutdown risk would be important. Each of these questions and the licensee's responses are presented in the following subsection, followed by the staff's findings regarding the impact of extended power uprate on shutdown operations.

8.3.1 Licensee Evaluation of Shutdown Operations

1. Does the application introduce new initiating events or change the frequencies of existing events?

The licensee stated that the possible initiating events during shutdown are generally defined as the loss of the shutdown safety functions, which include: decay heat removal, RCS inventory control, vital AC and DC power control, reactivity control, and containment closure. The licensee further stated that the extended power uprate does not increase the frequency of these initiators. Also, as discussed in Section 8.1.1 of this safety evaluation, the licensee indicated that no new initiating events or changes in the frequencies of existing initiating events are expected under extended power uprate conditions.

2. Does the application affect the scheduling of outage activities?

The licensee stated that the extended power uprate submittal indicates that the ability of the shutdown cooling system to achieve cold shutdown (i.e., less than 200 °F) in 36 hours has been verified and that the shutdown cooling system remains adequate to maintain refueling temperatures and a uniform boron concentration in the RCS.

Since the decay heat levels are expected to be slightly higher at extended power uprate conditions, the licensee stated that it may take a few hours longer to achieve cold shutdown. However, they concluded that this will cause very little change in the shutdown schedule and has no direct safety impacts on the schedule.

3. Does the application affect the ability of the operator to respond to shutdown events?

Though the extended power uprate does not increase the frequency of any initiators, the licensee does state that the extended power uprate may impact the operators' ability to respond to the loss of shutdown safety functions. Each of these shutdown safety functions are specifically addressed in the licensee's supplemental submittals and determined by the licensee to be adequate under extended power uprate conditions. This discussion includes examples from the licensee's shutdown operations protection plan for the most recent refueling outage (2R14) to demonstrate the various controls placed on the work performed during a typical refueling outage.

For nearly every shutdown safety function, the licensee indicated that the increase in RCS temperature and the increase in decay heat will decrease the time for the operators to respond to a loss of each shutdown safety function. However, the licensee claimed that maintaining adequate defense-in-depth for each safety function via their shutdown operations protection plan would minimize the impact of this decrease in response time.

Based on the licensee's supplemental submittal, the staff requested additional information on the shortest time to boiling for a typical outage and a description of any typical shutdown operations in which the containment might not be able to be closed within the estimated time to boiling. In response, the licensee indicated that the shortest time to boiling following entry into cold shutdown conditions during a typical outage is approximately 20 minutes. This shortest time is most likely to occur during the first reduced inventory window (i.e., during mid-loop operations). However, the licensee stated that typical shutdown operations never result in a containment closure time that exceeds the estimated time to boiling, even during mid-loop operations. Of all the containment breaches that usually occur during an outage, closure of the equipment hatch is the most limiting in terms of the amount of time required. The licensee stated that in numerous tests, ANO-2 has demonstrated its ability to achieve equipment hatch closure in 5 minutes to 15 minutes; usually in less than 10 minutes. Even so, during mid-loop operations, the containment equipment hatch is typically closed. For other breaches during this time, the ANO-2 outage risk management guidelines require that closure materials be staged in advance and when possible, closure capability be established from the outside. A person capable of quickly closing the flow path through the penetration must also be present. These actions assure that any breach of the containment will be closed well in advance of any boiling should a loss of shutdown cooling occur.

The licensee indicated that the outage risk management guidelines also state that during reduced inventory conditions, the only containment breaches allowed without specific approval of the operations manager are local leak rate test openings and via the containment ventilation/purge system. The outage risk management guidelines require that all containment breaches have the capability to be closed within 45 minutes and, where possible, within the estimated time to boiling. The 45-minute closure time is based on the guidance in NRC GL 88-17, "Loss of Decay Heat Removal." However, in recognition that the containment could become uninhabitable very quickly after the onset of boiling, the licensee stated that they make every effort to ensure containment closure can be completed in less than the estimated time to boiling. Further, the shutdown operations protection plan is developed for each outage based on the outage risk management guidelines. This plan identifies the minimum set of "safety functions/systems" required for various expected plant conditions during the outage. One of the safety functions addressed in the shutdown operations protection plan is containment closure.

The outage schedule is then reviewed against the requirements of the outage risk management guidelines and shutdown operations protection plan to ensure all requirements are met, including minimizing containment breaches while fuel is in the reactor, and ensuring the capability to close containment prior to the estimated time to boiling for those breaches that are scheduled. Thus, while the outage risk management guidelines allow for a containment breach that cannot be closed prior to the estimated time to boiling, such a breach is not considered in the outage schedule and would only occur if a gross penetration failure is found while at reduced inventory. The licensee stated that even then, the operations manager would have to be convinced that acceptance of the temporary condition is prudent, versus exiting the reduced inventory condition, after weighing all plant conditions at the time.

While decay heat will increase due to the extended power uprate, the above guidelines and philosophy for managing ANO-2 outages will not change. That is, the time to boiling at any given time following plant shutdown will decrease slightly following extended power uprate, compared to the current licensed power level, however, ANO-2 will continue to plan its outages to ensure that containment breaches can be closed prior to the estimated time to boiling.

4. Does the application affect the reliability or availability of equipment used for shutdown conditions?

As discussed previously in Section 8.1.2 of this safety evaluation, the licensee relies upon existing component monitoring programs to account for any additional wear that may result under extended power uprate conditions. The licensee stated that while some components may need to be refurbished or replaced more frequently due to wear under extended power uprate conditions, the functionality and reliability of the components will be maintained to the current standard.

5. Does the application affect the availability of equipment or instrumentation used for contingency plans?

The response to this question is the same as that for Question 4 above. The licensee relies upon existing component monitoring programs to account for any additional wear that may result under extended power uprate conditions. The licensee stated that while some components may need to be refurbished or replaced more frequently due to wear under extended power uprate conditions, the functionality and reliability of the components will be maintained to the current standard.

Based upon the above responses to the SRP 19.0 questions on shutdown risk, the licensee stated that they expect that the increase in decay heat will result in a proportionally small decrease in the time available for operator actions during shutdown operations. However, maintaining an adequate defense-in-depth for the shutdown safety functions at all times via the shutdown operations protection plan minimizes the impact of this decreased response time. Thus, the licensee concluded that the extended power uprate will have no unique or significant impacts on shutdown risk.

8.3.2 Staff Findings Regarding Impacts of Extended Power Uprate on Shutdown Operations

In SECY 97-168, “Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation,” the staff provided two estimates of PWR shutdown risk, which credited equipment required by TSs and equipment recommended to be available based on guidance from GL 88-17 and NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management.” These two "voluntary action cases" represent different interpretations of NUMARC 91-06 and GL 88-17. These two cases were not meant to bound plant operations, but were intended to be examples of reasonable interpretations of industry guidance. These two cases cover cold shutdown operations and refueling operations until the refueling cavity is flooded. Reduced inventory operations are a subset of this condition.

The high CDF voluntary action case represents a minimal level of implementation of both guidance documents in terms of the amount of extra equipment and additional sources of water being made available. For PWRs, the higher CDF voluntary action case includes the equipment credited by TSs, based on Westinghouse standard TSs, plus one ECCS pump, gravity feed, and an "available" containment. An "available" containment is defined as one that can be closed by remote or local manual actions before containment conditions become intolerable. The generic PWR high case had a CDF estimate of 8E-5/year.

The low CDF voluntary action case represents a more in-depth implementation of both guidance documents. The lower CDF case adds an additional emergency DG or equivalent power source, a second ECCS pump, containment spray pumps to supplement the residual heat removal pumps, and an enhanced recirculation capability. The generic PWR low case had a CDF estimate of 2E-6/year

Based on the ANO-2 shutdown cooling control procedures, the operators should have an HPSI flow path available at all times unless the RV is defueled. During reduced inventory operations, the licensee maintains a second flow path in addition to the HPSI flow path. However, based on conversations with the licensee, the second flow path may be a small charging pump that may not have the capability to keep the core covered following a loss of inventory event that includes a loss of both the residual heat removal flow path, which is the normal means of decay heat removal, and the HPSI flow path.

Concerning the licensee's containment closure capability, the outage risk management guidelines allow for a containment breach that cannot be closed prior to the estimated time to boiling. However, the licensee maintains that such a breach would not be incorporated into the outage schedule and, based on discussions with the licensee, such breaches would be unanticipated and/or inadvertent. The small increase in decay heat due to the proposed extended power uprate will reduce the time available for operator actions, such as to achieve containment closure. However, even for the most time-limiting closure action (i.e., the equipment hatch), which the licensee has demonstrated a closure capability of within 5 minutes to 15 minutes, the estimated time to boiling would be greater than 18 minutes for extended power uprate conditions as opposed to over 20 minutes for the pre-extended power uprate conditions. Therefore, the operator’s ability to inject before core damage and the ability to close containment before boiling should not be significantly changed, since: 1) there is margin between the time-limiting actions and the time to boiling, 2) the operators regularly calculate the time to boiling, and 3) the licensee maintains the availability of the core exit thermocouples to monitor RCS temperature until preparations for vessel head removal are complete.

Based on the staff’s review of the ANO-2 shutdown mitigation capability as communicated by the licensee’s responses, dated June 28, October 12, and December 20, 2001, to the staff’s

RAIs, the shutdown mitigation capability at ANO-2 appears to be closer to the high CDF voluntary action case. However, based on the information provided by the licensee, the staff was not able to provide a defensible quantitative estimate of the change in risk due to shutdown operations between the current and proposed uprate conditions, though it is expected to be small.

The licensee's risk evaluation for the extended power uprate did not include the quantification of the risks associated with shutdown operations and the risk impacts due to the extended power uprate on these operations (e.g., shorter operator response times). However, based on the licensee's description of their shutdown management guidelines and approach, including additional clarifications on how they have traditionally implemented and interpreted these guidelines, the staff believes that more detailed analyses would not significantly alter the approach or operations of the licensee under extended power uprate conditions during shutdown activities. Therefore, the staff concludes that pursuing further resolution of the licensee's approach to controlling the risks associated with shutdown operations would not significantly alter the overall conclusions of this specific license application and thus, would not warrant denial of this license amendment.

8.4 Quality of Probabilistic Risk Assessment

The quality of the licensee's PRA used to support a license application should be commensurate with the role that the PRA results play in the utility's and staff's decision-making process and should be commensurate with the degree of rigor needed to provide a valid technical basis for the staff's decision. In this case, the licensee is not requesting relaxation of any deterministic requirements for the proposed extended power uprate, and the staff's approval is primarily based on the licensee meeting the current deterministic requirements, with the risk assessment providing confirmatory insights. The staff's evaluation of the licensee's submittal focused on the capability of the licensee's PRA to analyze the risks stemming from both the current, pre-uprate plant operations, and the extended power uprate conditions. The staff's evaluation did not involve an in-depth review of the licensee's PRA.

Therefore, to determine whether the PRA used in support of the license application is of sufficient quality, scope, and detail, the staff evaluated the information provided by the licensee in their application and considered the review findings on the original ANO-2 IPE and IPEEE, as well as the fact that the ANO-2 PRA is controlled and the supporting models and calculations have been internally independently reviewed. In addition, by procedure, all plant changes, including hardware and procedural changes, are periodically reviewed and prioritized in terms of their impact on the PRA model. These changes are incorporated into the model in a manner consistent with their priority. An industry peer review has not been conducted on the ANO-2 PRA, but one is planned for early 2002.

The NRC staff evaluation report on the ANO-2 IPE was issued in May 1997 and concluded that the licensee had met the intent of GL 88-20. The licensee stated in their supplemental information associated with the ANO-2 extended power uprate license application that they have updated the PRA several times since the staff review relative to GL 88-20 to maintain it consistent with the as-built plant. The licensee stated that design engineering calculations and reports document the development of all major elements of the initial and updated versions of the PRA models, and that these calculations have been internally independently reviewed and are retained as quality records for the life of the plant. Similarly, the PRA evaluation performed in support of the ANO-2 extended power uprate has been internally independently reviewed by the licensee, and the specific calculations and reports will be retained as quality records for the life of the plant.

The NRC staff evaluation report on the ANO-2 IPEEE was completed in February 2001 and concluded, based on the staff's screening review, that the licensee's process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that ANO-2 had met the intent of Supplement 4 to GL 88-20.

Based on the NRC staff evaluation report on the ANO-2 IPEEE, the licensee stated that the overall plant high confidence of a low probability of failure capacity would be equal to or greater than the review level earthquake value of 0.3g after the resolution of certain outliers. In response to a staff RAI, the licensee stated that all outliers for ANO-2 have been resolved in accordance with their proposed resolutions, as presented in their IPEEE, and that all of their relied-upon equipment meets a minimum review level earthquake of 0.3g. Further, the extended power uprate does not modify the safe shutdown pathway assumed in the IPEEE seismic margins assessment. In addition, the licensee stated in their supplemental submittal that the RCS seismic analysis determined the dynamic response of the RCS with RSGs, considering system operating parameters consistent with the extended power uprate, and found the original analysis to remain bounding for the RSG configuration. Therefore, the seismic margins assessment performed for the IPEEE appears to be applicable for the extended power uprate plant conditions.

For the IPEEE fire analysis, the licensee's analysis is based on the EPRI fire-induced vulnerability evaluation methodology, which is oriented toward uncovering limiting plant design or operating characteristics (i.e., vulnerabilities) that make certain fire-initiated events more likely than others. As employed at ANO-2, the licensee asserted that the fire-induced vulnerability evaluation methodology is not a fire PRA, but a fire vulnerability analysis and as such, that it produces a conservatively high screening estimate for the CDF for each fire zone. The approach used by the licensee to evaluate the potential for fires is discussed in their submittal and are summarized above in Section 8.2.2. Based on the licensee's approach for evaluating the risk impact of fires due to extended power uprate conditions, which involved recalculating the pre- and post-extended power uprate fire zone contributions using the original IPEEE fire cutsets, the fire analysis appears to be appropriate for its use in identifying potentially new vulnerabilities and for conservatively assessing the impact of the extended power uprate from fires.

For the IPEEE evaluation of high winds, floods, and other external events, the licensee used the progressive screening approach described in NUREG-1407 and addressed in their IPEEE the aspects of their plant design that do not conform to the criteria provided in the 1975 SRP. Since the extended power uprate does not affect the high winds, floods, and other external events analysis, the IPEEE evaluation appears to be applicable to the extended power uprate conditions.

The staff finds that the ANO-2 internal events PRA is controlled and documented to ensure that it reflects the as-built, as-operated plant. With the completion of the plant modifications, in accordance with their proposed resolutions to address the previously identified seismic outliers, the staff finds that the ANO-2 IPEEE seismic margins assessment is applicable to the current and extended power uprate conditions. Further, the high winds, floods, and other external events were screened out for both the pre- and post-extended power uprate conditions and therefore, the staff finds that the IPEEE evaluation appears to be applicable to the extended power uprate conditions.

The licensee's risk evaluation for the extended power uprate did not include additional confirmatory analyses of the fire zones to ensure that the fire analysis more accurately reflects the extended power uprate plant conditions. However, based on the licensee's description of their fire analysis, including the conservatisms in their analysis, and the staff's site review of this analysis, the staff believes that further resolution of the fire analysis would not significantly alter the findings and operations of the licensee under extended power uprate conditions. Therefore, further resolution of this issue would not significantly alter the overall conclusions of this specific license application and thus, would not warrant denial of this license amendment. Therefore, the staff concludes that the ANO-2 internal and external events analyses are acceptable for this license application.

8.5 Probabilistic Risk Assessment Summary

The staff finds that no new impacts are expected for initiating event frequencies or component reliability, but potential impacts of the extended power uprate have been identified by the licensee for the success criteria of HPSI in the recirculation mode for LBLOCA events and for selected operator actions due to the decrease in available operator response times. The staff finds that the risk increases due to these impacts under the extended power uprate conditions are small and within the acceptance guidelines of RG 1.174.

The staff finds that the licensee has a process for managing plant risk during shutdown operations and that the risk impact due to the extended power uprate during these operations are expected to be small. The staff also finds that the risk increases from external events under extended power uprate conditions are expected to be small and within the acceptance guidelines of RG 1.174.

In conclusion, during the course of its review, the staff identified a few issues associated with the extended power uprate supporting risk analysis; predominately the conservatisms in the fire analysis. However, these issues do not raise questions regarding the licensee's ability to provide adequate protection by meeting deterministic requirements and regulations. Even though the estimated impacts due to extended power uprate are calculated conservatively, only the fire CDF slightly exceeds the RG 1.174 acceptance guidelines. The staff believes that further resolution of these analyses would not significantly alter the findings and current practices of the licensee under extended power uprate conditions to manage these risks. Therefore, the staff concludes that the identified issues do not warrant denial of this license application.

9.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS

9.1 License Paragraph 2.C.(1)

The licensee proposed to increase the maximum authorized core power level from 2815 MWt to 3026 MWt. The licensee has provided the results and the supporting information regarding its reanalyses and evaluations, as discussed previously throughout this safety evaluation. The staff has reviewed the licensee's submittal and finds that ANO-2 can be operated safely at a core power level of 3026 MWt.

9.2 Technical Specification Definition 1.3

The licensee proposed to revise the definition of RTP from 2815 MWt to 3026 MWt. The licensee has provided the results and the supporting information regarding its reanalyses and evaluations, as discussed previously throughout this safety evaluation. The staff has reviewed the licensee's submittal and finds that ANO-2 can be operated safely at a core power level of 3026 MWt.

9.3 Technical Specification Table 2.2-1, Item 5, and Technical Specification Table 3.3-4, Items 1c and 5c

The licensee proposed to reduce the low pressurizer pressure setpoint for the RPS and the ESFAS from ≥ 1675 psia to ≥ 1650 psia, with allowable values reduced from ≥ 1643.9 psia to ≥ 1618.9 psia. The licensee stated that this reduction in setpoint is needed to account for the larger pressurizer pressure decrease caused by the increased temperature swing and reduced RCS pressures following a reactor trip. This setpoint is sufficiently reduced to prevent unnecessary ESFAS actuations following normal plant transients, and remains sufficiently above the values that the licensee used in its analyses to support the proposed power uprate. Specifically, the licensee's LOCA and steam line break analyses used minimum setpoint of 1400 psia for the low pressurizer pressure safety injection actuation signal. On this basis, the staff finds the proposed TS changes acceptable.

9.4 Technical Specification 3.1.3.1 and Technical Specification 3.1.3.6, Table 3.3-1, and Bases 3/4.1.3

These TSs and their associated Bases were modified/clarified to accommodate the CEA position requirements and to be consistent with the power uprate. The majority of the changes are designed to provide clarifications to the wording or formatting of the TSs. The remaining revisions implement conservative or restrictive changes, as well as implement operational enhancements that are supported by the power uprate safety analysis. The staff finds these modifications and enhancements to be acceptable.

9.5 Technical Specification Table 3.3-4, Item 6b

The licensee proposed to eliminate the redundant requirement for the RWT volume and leave the indicated level as the controlling requirement because the indicated level is used for calibration and surveillance monitoring of the setpoint. The staff considers this to be an administrative change and finds it acceptable.

9.6 Technical Specification 3.5.4 and Bases 3/4.5.4

The current TS 3.5.4 specifies the required ranges of the borated water volume and their equivalent indicated levels in the RWT. In its application dated December 19, 2000, the licensee proposed replacing the RWT required range of water volumes and the equivalent indicated level with a required range of water volume used in the safety analyses. The licensee will move the required RWT indicated level to plant procedures. This proposal did not provide sufficient information in the TSs for operator control and NRC enforcement of this safety requirement.

In response to the staff's concern regarding this proposed change, in a supplemental letter dated October 31, 2001, the licensee indicated that the reason for removing the indicated level from this TS was to minimize updates to this TS for adjustments in instrument uncertainties and other conversion factors. To allow for operator control and NRC enforcement of this safety requirement, the licensee then proposed that the required RWT indicated water level be stated in Bases Section 3/4.5.4, with the same values specified in the current TS 3.5.4 (between 100% and 91.7%). In a supplemental letter dated December 6, 2001, the licensee stated that the values specified in the Bases are controlled by 10 CFR 50.59. The staff finds this revised proposal acceptable.

9.7 Technical Specification Table 3.7-1, Figure 3.7-1, and Bases 3/4.7.1

The licensee's proposed change will decrease the allowable values for the high linear power level trip setpoint during operation with one or more MSSVs inoperable. Also, TS Figure 3.7-1 will be changed and retitled to decrease the maximum value of the high linear power level trip setpoint that is allowed for a given MTC during periods when a MSSV is inoperable. The licensee indicated that these changes are appropriate due to the results of the safety analysis for loss-of-load/turbine trip for power uprate. The changes to Figure 3.7-1 will provide a more clear description regarding the MTC versus high linear power level trip setpoints at different MSSV inoperable conditions. The staff finds the licensee's proposal acceptable because the changes are toward the conservative direction from the current TSs at ANO-2.

9.8 Technical Specification 6.9.5.1

TS 6.9.5.1 lists the analytical methods used to determine the core operating limits. The licensee's proposed change will update the approved status of Topical Report CENPD-132-P, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," and reflect the changes in ECCS evaluation methods being applied to ANO-2. This proposed change is editorial. The staff finds it acceptable.

9.9 Bases 3/4.2.1

The Bases for the LHR TS will be changed to add a sentence to indicate that this limitation also ensures fuel pin pressures will not exceeded their design limits. This modification is consistent with the plant design associated with power uprate. This is an editorial change and it is acceptable.

10.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

11.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft environmental assessment and finding of no significant impact was prepared and published in the *Federal Register* on March 19, 2002 (67 FR 12587). The draft environmental assessment provided a 30-day opportunity for public comment. No comments were received on the environmental assessment. The final environmental assessment was published in the *Federal Register* on April 24, 2002 (67 FR 20176). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

12.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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